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Partitioning and transmutation Current developments – 2013

A report from the Swedish reference group for P&T-research

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Preface

This report is written on behalf of the Swedish reference group for research on partitioning and transmutation. The reference group has been assembled by SKB and its members represent the teams that are active in this field at Swedish universities.

Summary

The research and development on methods for partitioning and transmutation (P&T) of long-lived radionuclides in spent nuclear fuel has attracted considerable interest during the past decades. The main objective of P&T is to eliminate or at least substantially reduce the amount of such long-lived radionuclides that has to go to a deep geological repository for final disposal.

The radionuclides of main interest (concern) are those of the transuranium elements. These elements are formed in a nuclear reactor by one or more neutron captures in uranium atoms which then by subsequent radioactive decay are transformed to neptunium, plutonium, americium or curium. Even small amounts of elements heavier than curium are formed but these are of minor interest in this context. A few fission products (technetium-99, iodine-129) may also be of some interest for transmutation.

The long-lived radionuclides can be transmuted to more short-lived or stable nuclides by the use of nuclear physics processes. In theory and on laboratory scale several such processes are possible. In practice so far only transmutation by irradiation with neutrons can be achieved in macroscopic scale. Neutrons can cause fission in the transuranium elements and this process will release a substantial amount of energy. Thus transmutation on large scale of the transuranium elements from spent nuclear fuel must be done in a device similar to a nuclear reactor.

A prerequisite for transmutation by irradiation with neutrons is that the nuclides to be transmuted are separated (partitioned) from the other nuclides in the spent fuel. In particular the remaining uranium must be taken away unless production of more plutonium and other transuranium elements is desired. Separation of the various elements can, at least in principle, be achieved by mechanical and chemical processes. Currently there exist some large-scale facilities for separation of uranium and plutonium from the spent fuel – reprocessing plants. These do not, however, separate the minor actinides – neptunium[1], americium and curium – from the high level waste that goes to a repository. Plutonium constitutes about 90 % of the transuranium elements in fuel from light water reactors.

The objective of current research on *partitioning* is to find and develop processes suitable for separation of the heavier actinides (and possibly some long-lived fission products) on an industrial scale.

The objective of current research on *transmutation* is to define, investigate and develop facilities that may be suitable for transmutation of the aforementioned long-lived radionuclides.

The processes and facilities that could be implemented as results of such developments must meet very high standards of safety and radiation protection as well as have low environmental impact. They shall be economically viable and have good proliferation resistance. The large amount of energy released in the transmutation process should be used in a proper way. The processes and facilities must be acceptable to society.

Research on P&T started already in the 1950ies when development of nuclear power gained momentum. In the subsequent years it was mainly tied to the development of the breeder reactor. As the development of breeder reactors slowed down to a very low level in the early 1980ies the interest in P&T more or less disappeared.

The field experienced renewed interest through the 1990ies, however strongly focused on accelerator-driven hybrid systems (ADS) for incineration of spent nuclear fuel. The driving political rationale for this interest was that ADS was seen as a politically acceptable research topic in countries with a negative political nuclear landscape. Hence, it could serve as a training ground for future nuclear technology. Because ADS was the essentially only activity in the field for a period of more than a decade, the concept P&T became synonymous with ADS in the public debate. During the last few years, the interest in fast critical reactors for incineration of spent nuclear fuel has re-appeared on the scene. For instance, the joint European interest organization SNETP (Sustainable Nuclear Energy

[1] Note: Neptunium can be separated with uranium if a minor adjustment of the operating conditions is made in the industrial Purex process. This possibility is not used today as it would give increased costs for purification of recovered uranium.

Technology Platform) issued a strategic research agenda in 2008 in which the construction and commissioning of a fast reactor for transmutation is the highest priority, and design of such a reactor is in progress. Thus, now the interest in transmutation in critical reactors is larger than in ADS, and the research in ADS is to a large degree directed towards areas that are not technology-specific, i.e. where the results are useful regardless of whether critical or sub-critical accelerator-driven systems are employed.

In Europe the research has previously been focused on the R&D-programmes sponsored by the European Union (EU). The EU programmes have established a strong link between the various national programmes within the union and also with some other European countries. With the French initiative to construct the ASTRID reactor for industrial demonstration of the fast breeder technology, the dominance by the EU financed programmes might not prevail in the coming years. Other large programmes are going on in Japan, USA and Russia.

A review of the status of the efforts concerning P&T was published by SKB in 1998 (Enarsson et al. 1998). A second status report was compiled by the Swedish reference group on P&T-research in 2004 (Ahlström et al. 2004), a third similar report was issued in 2007 (Ahlström et al. 2007), and the most recent report stems from 2010 (Blomgren et al. 2010). The present report summarises the progress in the field through the years 2010–2012. It also provides an overview of the work performed in Sweden on P&T during the period.

SKB has been the main sponsor of P&T research in Sweden since 1995 until the end of 2009. In October 2009, VR (Vetenskapsrådet, the Swedish Research Council) granted 36 MSEK to the GENIUS project (Generation-IV in Universities in Sweden), which is a project targeting next generation critical reactors. Thereby, the research activity level in Sweden increased dramatically just before the present reporting period begun. Soon thereafter, an even larger project on collaboration with France on GenIV development, as well as other nuclear technology areas, was launched, although it has not reached full volume until recently. Thus, the Swedish domestic support has been larger than the EU funding in the last three-year period. At present, it is not clear whether this strong domestic support will continue.

These supports have led to large leverage of the original SKB funding. For instance, the P&T research at nuclear chemistry at Chalmers comprises 7.5 MSEK per year whereof 2 MSEK from SKB. At Uppsala University, the leverage is even larger with a P&T research turnover of 8 MSEK annually, whereof 1 MSEK provided by SKB. Similar leverage is present also at KTH. Since 2012, however, SKB has reduced and then stopped its financial support to this area.

In this report an overview is given of ongoing studies internationally and in Sweden and recent results from the Swedish research activities are presented.

System studies of partitioning and transmutation

The previous status report (Blomgren et al. 2010) summarised the results from a number of systems studies of partitioning and transmutation. The studies had been conducted on national and on international bases in Europe and USA as well as within OECD/NEA. Follow-up and supplement of the studies have been made by a couple of new studies within OECD/NEA.

Research coordinated at the EU level continues and is supplemented by specific national projects. Coordination of the work take place in SNE-TP, which is a technology platform for proposing and coordinating research, development and demonstration of future nuclear systems in Europe. Based on a Vision report presented in 2008 a Strategic Research Agenda (SRA) was developed to guide the future research coordination in Europe in 2009. The SRA was updated in 2013 and involves much of the research on P&T, as well as research on improved light water reactor systems. An important component of SNE-TP is to ensure the availability of research infrastructure in Europe.

Transmutation

During the period covered by the previous status report (2008-10), the major European project specifically focusing on a transmutation system was the EUROTRANS project. This four-year project started in April 2005 with a budget of 43 M€ of which 23 M€ were contributed by the European commission. It had 47 participants from 14 countries of which 10 represented industries, 19 research centres and 17 universities in Europe. In addition, a number of transmutation-supporting projects were undertaken, each far smaller than EUROTRANS, but in total they comprised a research volume comparable to EUROTRANS.

The experiences from the research community of such a huge single project were generally not positive. Therefore, in the next research framework program the strategy was smaller projects and a correspondingly larger number. The scope of the research did, however, not change dramatically. Still the major research areas have been system design, materials, fuel development and nuclear data. Coupling of an accelerator to a sub-critical core was an important theme in EUROTRANS, but has got less attention in the last few years, in line with the reduced interest in ADS.

The most import European projects with connection to transmutation performed during the period considered in this report are:

- ASTRID with the aim to construct and commission a 600 MWe sodium-cooled fast-neutron reactor in France, meeting highest level of safety and security standards. This is at present the most advanced project in Europe with connection to transmutation research.
- MYRRHA with the aim to construct and commission an Accelerator Driven System in Belgium with a lead cooled subcritical reactor. The objectives are to demonstrate the ADS concept and the feasibility of transmutation of minor actinides and long-lived fission products.
- CDT. The Central Design Team to design a fast spectrum transmutation experimental facility, supporting the core design for MYRRHA.
- GETMAT. Within the GETMAT project, structural materials for the primary circuit of Generation IV reactors are investigated.
- LEADER project should carry out the design of a lead-cooled fast reactor demonstrator, showing that the technology is capable for producing electricity at commercial conditions.
- ALFRED. Further development work and organization for the future construction of an Advanced Lead Fast Reactor European Demonstrator.
- PELGRIMM. Transmutation of minor actinides in oxide fuels and blankets, mainly according to the homogeneous and heterogeneous transmutation approaches studied in the French program.
- MAXIMA providing studies for the safety assessment of MYRRHA.
- ANDES will provide measurements and research to improve the accuracy and validation of nuclear data, in particular for the major actinides and new isotopes.
- ERINDA aims at providing a convenient platform to integrate all scientific efforts needed for high-quality nuclear data measurements in support of waste transmutation and Generation IV systems.
- ESFR has the objective to develop core design that optimizes performance, safety and fuel cycle issues for a sodium fast reactor.
- ARCAS aims at comparing, on a technological and economical basis, Accelerator Driven Systems and Fast Reactors as Minor Actinide Burners.

Swedish universities and research organizations participate in several of the above mentioned transmutation related European projects with assistance of the funding provided by Vetenskapsrådet, SKB and the Swedish – French collaboration. Also pure national projects are supported. The most important of the Swedish research connected to transmutation are:

- GENIUS, which is a consortium of KTH, Chalmers and Uppsala University supported by Vetenskapsrådet and with the focus on technology for lead cooled Generation IV reactors. Research activities are split in three major work packages:
  - Fuels with a laboratory at KTH for manufacture of uranium nitride (UN) and uranium-zirconium nitride (U,Zr)N fuels.
  - Materials for studies of special steels with high corrosion and radiation resistance in the environment of a lead cooled reactor.
  - Safety, looking at how uncertainties in nuclear data measurements propagate through construction of nuclear data libraries into neutronics simulations of safety parameters for lead cooled fast reactors.

- Swedish contributions to ASTRID. Two of the five work packages in the Swedish-French collaboration project are directly related to the design and safety of the ASTRID reactor. These include studies of severe accidents and core physics, diagnostics and instrumentation for enhanced safety. In addition there is a work package on nuclear data research, in which the existing experimental setup in Uppsala will be installed at a new neutron-beam facility under development at GANIL in France.
- KTH reactor physics. In addition to participating in the projects mentioned above KTH is performing a project to study to what extent existing light water reactors can be used for minor actinide transmutation.
- Uppsala University carries out experimental work for measurements of nuclear data at high energies 96 and 175 MeV, primarily at the quasi-monoenergetic neutron beam at the The Svedberg Laboratory, but also abroad.
- Some studies on Fusion-driven systems for transmutation as a possible future development.

Partitioning

The research on partitioning is following two main routes – hydrochemical processes and pyrochemical processes. In Europe the efforts were coordinated within the EU-project ACSEPT (Actinide Recycling by Separation and Transmutation) that started in 2008 and was completed in 2012. The project had an overall budget of 23.8 M€, whereof the EC contributed 9.0 M€. The project was a so-called collaborative project within the seventh framework programme sponsored by the European commission. It constituted a continuation of the project EUROPART within the previous framework programme. ACSEPT was divided into four domains:

1. Hydrochemistry studying aqueous separation processes dedicated to either actinide or grouped actinide separation.
2. Pyrochemistry studying two selected pyrochemical separation processes.
3. Process development integrating the experimental studies with engineering and system studies to provide flow sheets and recommendation for future demonstrations at pilot level.
4. Education and training implementing a training and education programme to share knowledge among the partitioning community.

Sweden was represented by the nuclear chemistry group at Chalmers in Göteborg.

In 2012, the ASGARD project started. ASGARD is a 4 year project funded by the EC Euroatom programme with a total budget of 9.6 M€. The project is coordinated by Chalmers University of Technology in Göteborg, Sweden. It comprises four domains:

1. Education and training.
2. Oxide fules.
3. Nitride fuels.
4. Carbide fuels.

Whereas the ACSEPT project was focused on partitioning techniques, ASGARD is more directed towards fuel development. Partly, this has been a natural consequence of the progress made in partitioning research in the last decade. Partitioning makes little sense if the separated material cannot be used in manufacturing of fuels for transmutation, and with the rapid development in partitioning a few years ago, fuel production now constitutes a limiting factor for overall success.

It should be pointed out that the development seems to go back and forth between partitioning and fuel development, which is natural. Partitioning research has resulted in reasonably successful methods to separate raw material for fuels for transmutation. The next step is research aiming at developing high-performance fuels to be irradiated in fast reactors. If successful, this could re-direct the scientific focus back to partitioning, not of light-water reactor spent fuel but of the fuels with high trans-uranium content after transmutation. That would ultimately be needed to allow multiple, or even indefinite, re-cycling of fuels with high trans-uranium content, i.e., the ultimate dream of P&T.

In addition to ACSEPT and ASGARD mentioned above the most import European projects with connection to partitioning and fuel development performed during the period considered in this report are:

- ACTINET, which is a network to provide access for external users to specialised facilities for training and research on partitioning. The facilities are CEA (Atalante, DPG), ITU, KIT (INE labs and beamline), FZD (IRC, ROBL) and PSI (Macro XAS beamline).
- CINCH develops education material for nuclear chemistry and offers education at several levels of students (PhD, life-long learning and M.Sc.).
- SEARCH investigates the chemical behaviour of the fuel and coolant in MYRRHA in support of the licensing process.
- EURACT-NMR is a support action to make access possible to European nuclear licensed facilities with advanced nuclear magnetic resonance spectrometers.

Swedish universities and research organizations participate in several of the above mentioned European projects with assistance of the funding provided by Vetenskapsrådet and SKB. Also pure national projects are supported. The most important of the Swedish research connected to partitioning and fuel development are:

- Development of a GANEX process for liquid-liquid extraction of the actinide group.
- Diluent effects. Studies have been performed on alcohol diluents and the BTBP ligands, as well as silver co-extracts in support of the GANEX process development.
- Fuel lab. A fuel fabrication lab for transuranium elements has been installed to fabricate $(Pu,Zr)O_2$ and (Pu,Zr)N fuel.
- RADTox is a program for estimating the radiotoxicity with varying partitioning and transmutation effects.

Future P&T in Sweden

Since this report was compiled (2013) the economical support to P&T has been reduced substantially. SKB ended the P&T support by 2014 and support from the Swedish Research Council diminished by the same time. The consequence of this is that Swedish researchers have limited possibility to be engaged in the on-going European cooperation projects.

SKB's reason for ending the financing is that a geological (KBS-3) repository has been selected and the application for construction is now being scrutinized by the authorities. P&T is therefore no longer needed as an alternative option.

From a Swedish perspective it is important to follow the international development on new reactors and P&T and maintain a reasonable level of competence within the country. The competence developed by research on P&T is valuable not only for evaluating the progress and potential within this field but also for development of safety and fuel supply at existing nuclear facilities.

The application of P&T for an effective and substantial decrease in the amount of long-lived elements that need final disposal implies the use of nuclear energy during a very long time – beyond 100 years. Though there are legal possibilities for the power companies to plan for new reactors as exchange for the ones taken out of operation there are presently no plans for continuing nuclear power production after the present reactors have been taken out of operation.

A successful development of P&T will not eliminate the need for deep geologic repositories for high-level waste and for long-lived wastes. The complex processes will unavoidably create waste streams containing small amounts of long-lived radionuclides. The development may, however, decrease the demands on engineered barriers. It may also decrease the required volumes of high-level waste in the repositories (volumes of low- and intermediate-level waste will, on the other hand, tend to increase as a result of the partitioning processes).

Sammanfattning

Forskning och utveckling av metoder för separation och transmutation av långlivade radionuklider i använt kärnbränsle har väckt stort intresse under de senaste årtiondena. Huvudsyftet med separation och transmutation är att eliminera eller åtminstone väsentligt minska mängden långlivade radionuklider som måste slutförvaras i ett geologiskt djupförvar.

Radionukliderna av primärt intresse är transuranerna. Dessa grundämnen bildas i en kärnreaktor av en eller flera neutroninfångningar i uranatomer, som sedan genom efterföljande radioaktivt sönderfall omvandlas till neptunium, plutonium, americium eller curium. Små mängder av ännu tyngre ämnen än curium kan bildas, men de är mindre intressanta i detta sammanhang. Ett fåtal fissionsprodukter (teknetium-99, jod-129) kan också vara av visst intresse för transmutation.

De långlivade radionukliderna kan transmuteras till mer kortlivade eller stabila nuklider i kärnfysikaliska processer. Teoretiskt och i laboratorieskala är flera sådana processer möjliga. I praktiken kan hittills bara transmutation genom neutronbestrålning uppnås i makroskopisk skala. Neutroner kan orsaka fission i transuranerna och denna process frigör en betydande mängd energi. Därför måste transmutation av transuraner från använt kärnbränsle i stor skala utföras i en anordning som liknar en kärnreaktor.

En förutsättning för transmutation genom neutronbestrålning är att de nuklider som ska transmuteras separeras från andra nuklider. I synnerhet måste det återstående uranet avlägsnas, om inte produktion av mer plutonium och andra transuraner är önskvärd. Separation av olika ämnen kan, åtminstone i princip, uppnås genom mekaniska och kemiska processer. För närvarande finns det några anläggningar i stor skala för separation av uran och plutonium från använt kärnbränsle, så kallade upparbetningsanläggningar. De kan emellertid inte avskilja de mindre aktiniderna – neptunium[2], americium och curium – från högaktivt avfall som går till slutförvar. Plutonium utgör cirka 90 % av transuranerna i bränsle från lättvattenreaktorer.

Målet för den pågående forskningen om *separation* är att hitta och utveckla processer som är lämpliga för separation av tyngre aktinider (och eventuellt vissa långlivade fissionsprodukter) i industriell skala.

Målet för den pågående forskningen om *transmutation* är att definiera, undersöka och utveckla anläggningar som kan vara lämpliga för transmutation av de ovan nämnda långlivade radionukliderna.

De processer och anläggningar som skulle kunna implementeras som resultat av sådan utveckling måste uppfylla mycket höga krav på säkerhet och strålskydd samtidigt som de har låg miljöpåverkan. De ska vara ekonomiskt bärkraftiga och gynnsamma utifrån ett icke-spridnings perspektiv. Den stora mängden energi som frigörs i transmutationsprocessen bör också tas till vara. Processerna och anläggningarna måste accepteras av samhället.

Forskningen inom separation och transmutation inleddes redan på 50-talet när kärnkraftsutvecklingen tog fart. Efterföljande år var den främst inriktad på utveckling av bridreaktorer. När utvecklingen av bridreaktorer saktade ner till en mycket låg nivå under det tidiga 80-talet avtog intresset för separation och transmutation nästan helt.

Området väckte nytt intresse under 90-talet, dock med starkt fokus på acceleratordrivna hybridsystem (ADS) för förbränning av använt kärnbränsle. Det drivande politiska motivet för intresset var att ADS sågs som ett politiskt acceptabelt forskningsämne i länder med en negativ politisk inställning till kärnkraft. Därför skulle det kunna fungera som ett övningsfält för framtida kärnteknik. Eftersom ADS var den väsentligen enda aktiviteten i området under en period på mer än ett decennium, blev konceptet separation och transmutation synonymt med ADS i den offentliga debatten. Under 2000-talet har intresset för snabba kritiska reaktorer för förbränning av använt kärnbränsle väckts på nytt.

[2] Observera: Neptunium kan separeras med uran om en mindre justering av driftförhållandena görs i den industriella Purex-processen. Denna möjlighet används inte i dag, eftersom den skulle leda till ökade kostnader för rening av återvunnet uran.

Till exempel fastställde den gemensamma europeiska intresseorganisationen SNETP (Sustainable Nuclear Energy Technology Platform) en strategisk forskningsagenda år 2008, i vilken uppförande och driftsättning av en snabbreaktor för transmutation är av högsta prioritet, och design av en sådan reaktor pågår. Därför är intresset för transmutation i kritiska reaktorer nu större än för ADS och forskningen inom ADS är i hög grad inriktad på områden som inte är teknikspecifika, dvs. där resultaten är användbara oavsett om kritiska eller underkritiska acceleratordrivna system används.

I Europa har forskningen tidigare varit inriktad på FoU-program sponsrade av EU. EU-programmen har skapat en stark koppling mellan de olika nationella programmen inom unionen och också med vissa andra europeiska länder. Med det franska initiativet att konstruera ASTRID-reaktorn för industriell demonstration av den snabba bridtekniken, är det möjligt att de EU-finansierade programmens dominans inte fortsätter under kommande år. Andra stora program pågår i Japan, USA och Ryssland.

En genomgång av statusen för insatser rörande separation och transmutation publicerades av SKB 1998 (Enarsson et al. 1998). En andra statusrapport sammanställdes av den svenska referensgruppen för forskning om separation och transmutation 2004 (Ahlström et al. 2004), en tredje liknande rapport 2007 (Ahlström et al. 2007) och den senaste rapporten är från 2010 (Blomgren et al. 2010). Denna rapport sammanfattar utvecklingen inom området under åren 2010–2012. Den ger också en översikt av det arbete som utförts i Sverige kring separation och transmutation under perioden.

SKB har varit huvudsponsor för forskningen om separation och transmutation i Sverige från 1995 fram till slutet av 2009. I oktober 2009 beviljade VR (Vetenskapsrådet) 36 MSEK till projektet GENIUS (Generation IV-forskning i universitets-Sverige), som är ett projekt med inriktning på nästa generation av kritiska reaktorer. Därmed ökade aktivitetsnivån för forskningen i Sverige dramatiskt strax innan den nuvarande rapportperioden inleddes. Kort därefter lanserades ett ännu större projekt i samarbete med Frankrike om GenIV-utveckling och andra kärntekniska områden, även om det inte uppnådde full volym förrän nyligen. Således har det svenska inhemska stödet varit större än EU:s finansiering under den senaste treårsperioden. För närvarande är det inte klart om detta starka inhemska stöd kommer att fortsätta.

Dessa stöd har lett till stora förstärkningseffekter för den ursprungliga SKB-finansieringen. Till exempel omfattar forskningen om separation och transmutation vid kärnkemi på Chalmers 7,5 MSEK per år, varav två MSEK från SKB. Vid Uppsala universitet är förstärkningseffekten ännu större med en omsättning för forskning om separation och transmutation på 8 MSEK årligen, varav 1 MSEK finansieras av SKB. Liknande förstärkningseffekter finns också på KTH. Sedan 2012 har SKB dock minskat och sedan helt avslutat sitt ekonomiska stöd till detta område.

I den här rapporten ges en översikt av pågående studier internationellt och i Sverige och aktuella resultat från den svenska forskningsverksamheten redovisas.

Systemstudier av separation och transmutation

Föregående statusrapport (Blomgren et al. 2010) sammanfattade resultaten från ett antal systemstudier av separation och transmutation. Studierna hade utförts nationellt och internationellt i Europa och USA samt inom OECD/NEA. Uppföljning och komplettering av studierna har gjorts genom några nya studier inom OECD/NEA.

Forskning som samordnas på EU-nivå fortsätter och kompletteras med specifika nationella projekt. Samordning av arbetet görs i SNE-TP, som är en teknologiplattform för att föreslå och samordna forskning, utveckling och demonstration av framtida kärntekniska system i Europa. Baserat på Visionrapporten som presenterades 2008 togs en strategisk forskningsagenda (SRA) fram 2009 för att vägleda framtida samordning av forskning i Europa. Forskningsagendan uppdaterades 2013 och omfattar en stor del av forskningen på separation och transmutation, liksom forskning kring förbättrade lättvattenreaktorsystem. En viktig komponent i SNE-TP är att säkerställa tillgången på forskningsinfrastruktur i Europa.

Transmutation

Under tidsperioden som täcks av föregående statusrapport (2008-10) var det största europeiska projektet specifikt inriktat på ett transmutationssystem EUROTRANS-projektet. Det fyraåriga projektet inleddes i april 2005 med en budget på 43 M €, varav 23 M € var bidrag från Europeiska kommissionen. Det hade 47 deltagarna från 14 länder, av vilka 10 representerade industrin, 19 forskningscentra och

17 universitet i Europa. Utöver detta har ett antal projekt kring transmutation genomförts, vart och ett mycket mindre än EUROTRANS, men tillsammans utgjorde de en forskningsvolym som är jämförbar med EUROTRANS.

Forskarvärldens erfarenheter av ett sådant enormt enskilt projekt var i allmänhet inte positiva. Därför var strategin i nästa ramprogram för forskning mindre projekt och ett motsvarande större antal. Forskningens omfattning förändrades emellertid inte markant. De stora forskningsområdena har alltjämt varit systemkonstruktion, material, bränsleutveckling och kärndata. Koppling av en accelerator till en underkritisk reaktorhärd var ett viktigt tema i EUROTRANS, men har blivit mindre uppmärksammat under de senaste åren, i takt med det minskade intresset för ADS.

De viktigaste europeiska projekten kopplade till transmutation som genomförts under den tidsperiod som beaktas i denna rapport är:

- ASTRID, med syfte att uppföra och driftsätta en 600 MWe natriumkyld snabbneutronreaktor i Frankrike, som uppfyller den högsta nivån för säkerhet och säkerhetsstandarder. Detta är för närvarande det mest avancerade projektet i Europa kopplat till transmutationsforskning.
- MYRRHA, med syfte att uppföra och driftsätta ett acceleratordrivet system i Belgien med en blykyld underkritisk reaktor. Målet är att demonstrera ADS-konceptet och genomförbarheten för transmutation av mindre aktinider och långlivade fissionsprodukter.
- CDT. Centralt designteam (Central Design Team) för att utforma en experimentell anläggning för snabbspektrumtransmutation, som stöd för härdkonstruktionen i MYRRHA.
- GETMAT. Inom GETMAT-projektet undersöks strukturella material för primärkretsen i fjärde generationens reaktorer.
- LEADER-projektet ska utforma en demonstrationsanläggning för en blykyld snabbreaktor, som visar att tekniken kan producera el under kommersiella förhållanden.
- ALFRED. Fortsatt utveckling och organisation för framtida konstruktion av en europeisk demonstrationsanläggning för en avancerad blykyld snabbreaktor (Advanced Lead Fast Reactor European Demonstrator).
- PELGRIMM. Transmutation av mindre aktinider i oxidbränslen och täcken, huvudsakligen i enlighet med de homogena och heterogena tillvägagångssätt för transmutation som studerats i det franska programmet.
- MAXIMA utför studier för säkerhetsanalysen av MYRRHA.
- ANDES ska tillhandahålla mätningar och forskning för att förbättra noggrannhet och validering av kärndata, i synnerhet för de tyngre aktiniderna och nya isotoper.
- ERINDA syftar till att tillhandahålla en lämplig plattform för att integrera alla vetenskapliga insatser som behövs för högkvalitativa kärndatamätningar som stöd för avfallstransmutation och fjärde generationens system.
- ESFR har som mål att utveckla en härdkonstruktion som optimerar prestanda, säkerhet och bränslecykelfrågor för en natriumsnabbreaktor.
- ARCAS syftar till att jämföra, på teknisk och ekonomisk grund, acceleratordrivna system och snabbreaktorer ur perspektivet förbränning av de minder aktiniderna.

Svenska universitet och forskningsorganisationer har deltagit i flera av de ovan nämnda transmutationsrelaterade europeiska projekten med stöd av finansiering från Vetenskapsrådet, SKB och det svensk-franska samarbetet. Även rent nationella projekt får stöd. Den viktigaste svenska forskningen kopplad till transmutation är:

- GENIUS, som är ett konsortium av KTH, Chalmers och Uppsala universitet, med stöd från Vetenskapsrådet och med fokus på teknik för blykylda Generation IV-reaktorer. Forskningsinsatserna är uppdelade i tre större arbetspaket:
  - Bränslen, med ett laboratorium vid KTH för tillverkning av urannitrid(UN)- och uranzirkoniumnitrid(U,Zr)N-bränslen.
  - Material, för studier av specialstål med stor motståndskraft mot korrosion och strålning i den miljö som utgörs av en blykyld reaktor.
  - Säkerhet, tittar på hur osäkerheter i kärndatamätningar propagerar genom att skapa kärndatabibliotek i neutroniksimuleringar av säkerhetsparametrar för blykylda snabbreaktorer.

- Svenska bidrag till ASTRID. Två av de fem arbetspaketen i det svensk-franska samarbetsprojektet är direkt relaterade till utformning och säkerhet för ASTRID-reaktorn. Dessa omfattar studier av allvarliga olyckor och härdfysik, diagnostik och instrumentering för ökad säkerhet. Dessutom finns det ett arbetspaket om kärndataforskning, i vilket den befintliga experimentuppställningen i Uppsala kommer att installeras i en ny neutronstråleanläggning som är under utveckling vid GANIL i Frankrike.
- KTH reaktorfysik. Utöver att delta i de projekt som nämns ovan genomför KTH ett projekt för att studera i vilken utsträckning befintliga lättvattenreaktorer kan användas för transmutation av mindre aktinider.
- Uppsala universitet genomför experimentellt arbete för mätningar av kärndata vid hög energi, 96 och 175 MeV, i första hand med den kvasimonoenergetiska neutronstrålen i Svedberg-laboratoriet, men även utomlands.
- Några studier om fusionsdrivna system för transmutation som en möjlig framtida utveckling.

Separation

Forskningen om separation följer två huvudspår – hydrokemiska processer och pyrokemiska processer. I Europa samordnades insatserna inom EU-projektet ACSEPT (Actinide Recycling by Separation and Transmutation) som inleddes 2008 och avslutades 2012. Projektet hade en total budget på 23,8 M €, varav EU-kommissionen bidrog med 9,0 M €. Projektet var ett så kallat samarbetsprojekt inom det sjunde ramprogrammet sponsrat av Europeiska kommissionen. Det utgjorde en fortsättning av projektet EUROPART inom det tidigare ramprogrammet. ACSEPT delades in i fyra domäner:

1. Vattenkemi, som studerar vattenbaserade separationsprocesser för antingen aktinider eller grupperad aktinidseparation.
2. Pyrokemi, som studerar två utvalda pyrokemiska separationsprocesser.
3. Processutveckling, som integrerar experimentella studier med teknik- och systemstudier för att tillhandahålla flödesscheman och rekommendationer för framtida demonstrationer på pilotnivå.
4. Utbildning och träning, som genomför ett tränings- och utbildningsprogram för att dela kunskap i forskarsamhället kring separation.

Sverige representerades av kärnkemigruppen vid Chalmers i Göteborg.

Under 2012 startade ASGARD-projektet. ASGARD är ett fyraårigt projekt som finansieras av EU-kommissionens Euratom-program med en total budget på 9,6 M €. Projektet samordnas av Chalmers tekniska högskola i Göteborg, Sverige. Det omfattar fyra domäner:

1. Utbildning och träning.
2. Oxidbränslen.
3. Nitridbränslen.
4. Karbidbränslen.

Medan ACSEPT-projektet fokuserade på separationstekniker, är ASGARD mer inriktat på bränsleutveckling. Delvis är det en naturlig konsekvens av framstegen inom separationsforskningen det senaste decenniet. Separation är inte meningsfull om inte det separerade materialet kan användas vid tillverkning av bränslen för transmutation, och i och med den snabba utvecklingen inom separation för ett par år sedan utgör bränsleproduktion nu den begränsande faktorn för allmän framgång.

Det bör poängteras att utvecklingen verkar gå fram och tillbaka mellan separation och bränsleutveckling, vilket är naturligt. Separationsforskningen har resulterat i någorlunda framgångsrika metoder för att separera råmaterial till bränslen för transmutation. Nästa steg är forskning med syfte att utveckla högeffektiva bränslen att bestråla i snabbreaktorer. Om det lyckas skulle vetenskapligt fokus kunna skifta tillbaka till separation, inte av använt kärnbränsle från lättvattenreaktorer, utan av bränslen med högt transuraninnehåll efter transmutation. Det skulle krävas i slutändan för att medge multipel, eller till och med obegränsad, återvinning av bränslen med högt transuraninnehåll, det vill säga den ultimata drömmen för separation och transmutation.

Förutom ACSEPT och ASGARD som nämns ovan är de viktigaste europeiska projekten knutna till separation och bränsleutveckling under den tidsperiod som beaktas i denna rapport:

- ACTINET, som är ett nätverk för att ge externa användare tillgång till specialiserade anläggningar för utbildning och forskning på separation. Anläggningarna är CEA (Atalante, DPG), ITU, KIT (INE labs and beamline), FZD (IRC, ROBL) och PSI (Macro XAS beamline).
- CINCH tar fram utbildningsmaterial för kärnkemi och erbjuder utbildning för studenter på flera olika nivåer (PhD, livslångt lärande och M.Sc.).
- SEARCH undersöker det kemiska beteendet hos bränsle och kylmedel i MYRRHA som stöd för tillståndsprövningen.
- EURACT-NMR är en stödverksamhet för att möjliggöra åtkomst till europeiska kärntekniska anläggningar med avancerade kärnmagnetiska resonansspektrometrar.

Svenska universitet och forskningsorganisationer har deltagit/deltar i flera av de ovan nämnda europeiska projekten med hjälp av finansiering från Vetenskapsrådet och SKB. Även rent nationella projekt får stöd. Den viktigaste svenska forskningen kopplad till separation och bränsleutveckling är:

- Utveckling av en GANEX-process för vätskeextraktion av aktinider.
- Utspädningseffekter. Studier har gjorts på alkoholspädningsmedel och BTBP-ligander, liksom samextraktion av silver, som stöd för utvecklingen av GANEX-processen.
- Bränslelaboratorium. Ett bränsletillverkningslaboratorium för transuraner har installerats för att tillverka $(Pu,Zr)O_2$- och (Pu,Zr)N-bränsle.
- RADTox är ett program för att uppskatta radiotoxicitet med varierande separations- och transmutationseffekter.

Framtida separation och transmutation i Sverige

Sedan denna rapport sammanställdes (2013) har det ekonomiska stödet för separation och transmutation minskat väsentligt. SKB avslutade sitt stöd till separation och transmutation 2014 och stödet från Vetenskapsrådet minskade samtidigt. Konsekvensen är att svenska forskare har begränsade möjligheter att delta i pågående europeiska samarbetsprojekt.

SKB:s motiv för att avsluta finansieringen är att ett geologiskt (KBS-3) förvar har valts och tillståndsansökan för uppförandet granskas nu av myndigheterna. Separation och transmutation behövs därför inte längre som ett alternativ.

Från ett svenskt perspektiv är det viktigt att följa den internationella utvecklingen för nya reaktorer och separation och transmutation och upprätthålla en rimlig kompetensnivå inom landet. Kompetensen som utvecklas genom forskning på separation och transmutation är värdefull, inte bara för att utvärdera framsteg och möjligheter inom detta område, utan också för utveckling av säkerhet och bränsleförsörjning vid existerande kärntekniska anläggningar.

Tillämpning av separation och transmutation för en effektiv och väsentlig minskning av mängden långlivade ämnen som behöver slutförvaring förutsätter att kärnenergi används under mycket lång tid – längre än 100 år. Även om det finns rättsliga möjligheter för kärnkraftsföretagen att planera för nya reaktorer som utbyte för dem som tas ur drift finns för närvarande inga planer för fortsatt kärnkraftsproduktion efter att de nuvarande reaktorerna har avvecklats.

En framgångsrik utveckling för separation och transmutation kommer inte att eliminera behovet av geologiska djupförvar för högaktivt avfall och långlivat avfall. De komplexa processerna kommer oundvikligen att skapa avfallsströmmar med små mängder långlivade radionuklider. Utvecklingen kan dock minska kraven på tekniska barriärer. Den kan även minska volymerna av högaktivt avfall i slutförvaren (volymer av låg- och medelaktivt avfall kommer, å andra sidan, att öka som en följd av separationsprocesserna).

Contents

1 Introduction

The research and development on methods for partitioning and transmutation (P&T) of long-lived radionuclides in spent nuclear fuel has attracted considerable interest since the early 1990ies. The main objective of P&T is to eliminate or at least substantially reduce the amount of such long-lived radionuclides that has to go to a deep geological repository for final disposal.

The radionuclides of main interest (concern) are those of the transuranium elements – neptunium, plutonium, americium and curium. These elements are formed in a nuclear reactor by one or more neutron captures in uranium atoms and subsequent radioactive decay. The reduction of long-lived radionuclides can be achieved by transmutation of the nuclides by the use of nuclear physics processes. In theory, several such processes are possible. In practice so far only transmutation by irradiation with neutrons can be made in macroscopic scale. Neutrons can cause fission in the transuranium elements, hence transforming them into stable or more short-lived isotopes. This process will release a substantial amount of energy and thus transmutation on large scale of the transuranium elements from spent nuclear fuel must be done in a nuclear reactor.

A prerequisite for transmutation by irradiation with neutrons is that the nuclides to be transmuted are separated (partitioned) from the other nuclides in the spent fuel. In particular the remaining uranium must be taken away. Otherwise more plutonium and other transuranic elements are produced. Separation of the various elements can at least in principle be achieved by (mechanical and) chemical processes. Currently large scale facilities (reprocessing plants) exist for the separation of uranium and plutonium from the spent nuclear fuel. These facilities cannot separate the heavier transuranic elements – neptunium[3], americium and curium – these elements remain in the waste from reprocessing, which will be disposed of in a final repository for HLW (High Level Waste). Plutonium constitutes about 90 % of the transuranium elements in fuel from light water reactors.

The objective of current research on *partitioning* is to find and develop processes suitable for separation of the heavier minor actinides (and possibly some long-lived fission products) on an industrial scale.

The objective of current research on *transmutation* is to define, investigate and develop facilities that may be suitable for transmutation of the aforementioned long-lived radionuclides.

The processes and facilities that could be implemented as results of such developments must meet very high standards of safety and radiation protection as well as have low environmental impact. They shall be economically viable and have good proliferation resistance. The large amount of energy released in the transmutation process should be used in a proper way. In other words the processes and facilities must be acceptable to society.

Research on P&T started already in the 1950ies when development of nuclear power gained momentum. In the subsequent years it was mainly tied to the development of the breeder reactor. As this development slowed down in the Western world to a very low level in the early 1980ies the interest in P&T more or less disappeared.

The renewed interest through the 1990ies has caused some expansion of the programmes in this field in particular on an international level. In Europe this has until recently been focused on the R&D-programmes of the European Union (EU). The EU so-called framework programmes (FP) have established a strong link between the various national programmes within the union and also in some other European countries. Other large programmes are going on in Japan, USA and Russia.

The programme in Sweden has until recently mainly been financed by SKB and was initiated in the early 1990ies. At present research on partitioning is made by the nuclear chemistry group of

[3] Neptunium can be separated with uranium if a minor adjustment of the operating conditions is made. This possibility is not used today. In fact almost 30 % of the neptunium is, however, already recovered in the uranium purification part of the plants and then returned to the high level waste stream together with trace amounts of plutonium and uranium.

the department of chemical and biological engineering at Chalmers University of Technology in Göteborg. The research on transmutation is made at the departments of nuclear and reactor physics and of nuclear safety at the Royal Institute of Technology (KTH) in Stockholm. Research on basic physics data for transmutation is made at the division of applied nuclear physics at Uppsala University. All these Swedish research teams are participating in projects that are partly financed by the European Commission of EU. They are also engaged in other international cooperation.

Earlier reviews of the status of the efforts concerning P&T were published by SKB in 1998 (Enarsson et al. 1998), 2004, 2007 (Ahlström et al. 2004, 2007) and 2010 (Blomgren et al. 2010).

This current status report summarises the work reported in the years 2010–2012 and tries to assess the prospects for future development of P&T as seen from a Swedish perspective. The objectives of this report are to:

- Present the current situation on P&T.
- Fulfil the requirement concerning reporting about "current status" in the statutes for the aforementioned P&T reference group.

The report has been co-authored by members (and their associates) of a Swedish reference group on P&T-research established by SKB.

The report is structured in four main chapters. Chapter 2 gives a short account of recent international and foreign work concerning P&T. Chapter 3 is devoted to presentation of some results from the Swedish work. Chapter 4 summarizes some views on the need for future R&D important for P&T. The last chapter, Chapter 5 gives a short update of the assessments of the prospects for P&T presented in the 2010 status report (Blomgren et al. 2010).

2 Summary of some international studies since 2010

2.1 ASTRID

The objective of the integrated technology demonstrator ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) is to ensure industrial-scale demonstration of a Generation IV Sodium Fast Reactor, meeting the highest level of safety and security standards, and providing significant improvements in terms of industrial operation. The reactor is expected to operate around 2022. The ASTRID programme encompasses the ASTRID reactor itself, the realisation of sodium technological loops and the validation of components, as well as the construction of a fuel manufacturing workshop.

The planned construction site is Marcoule in the Rhône valley, about 30 km north of Avignon. ASTRID will be a pool type, sodium cooled fast reactor of 1 500 MWth, generating about 600 MWe. That level of power is required to guarantee representativeness in terms of design, operation and safety demonstration, of the reactor core and main components, and will compensate for the operational costs by generating a significant amount of electricity.

ASTRID will also have provisions for experiments on transmutation of minor actinides (mainly americium) in significant quantities to allow optimised management of wastes. Though future fast reactor plants intend to be breeders, ASTRID will be an iso-generator.

ASTRID will be equipped for experiments. Its design must therefore be flexible enough to be able to test innovative options that were not chosen for the initial design. Novel instrumentation technologies, new fuels and even new system components will be tested in ASTRID. It will be equipped with a hot cell for examining irradiation objects, built either in the plant or nearby.

Since September 2012, the following partners have been involved in the conceptual design phase:

- AREVA NP for nuclear island (core and fuel stays with CEA).
- EDF for support to the owner and contribution to R&D.
- ALSTOM for turbine island.
- BOUYGUES for civil engineering.
- COMEX NUCLEAIRE: innovative studies in robotics and mechanics.
- TOSHIBA for development of large electromagnetic pumps.
- JACOBS for infrastructures.
- ROLLS-ROYCE for research and technology development on sodium gas exchangers and fuel handling.
- ASTRIUM for reliability, maintainability and availability analysis.

The ASTRID design benefits from the advantages of pool-type, sodium-cooled fast neutron reactors, which provide very favourable inherent safety margins with regard to Fukushima-like events:

- The main vessel contains the whole primary system including the core, the intermediate heat exchangers and the primary pumps, giving pooltype SFR a high level resistance to loss of coolant accidents (LOCA), since it is possible to install a guard vessel around the main vessel. Furthermore, the primary system is not pressurised.
- The intermediate system (or secondary system) uses sodium loops to transfer the energy from the primary circuit to the main heat exchangers and provides an additional barrier.
- The size and mass of the primary system, along with the quantity of primary coolant and its physical properties provides for a very large thermal inertia of the reactor, the thermal inertia allowing larger grace times in order to put in operation the DHR systems.

The ASTRID core is an heterogeneous MOX core called CFV ("Coeur à Faible effet de Vide sodium" or "Low sodium Void Worth Core"), that is characterised by a sodium void coefficient close to 0.

This is a major difference from classical fast neutron reactor designs, and provides the ASTRID core with improved inherent behaviour in terms of safety.

An application was submitted to the French parliament in the autumn 2012 documenting the pre-design and requesting funding for a detailed design. At present writing, evaluation is in progress aiming at a decision early summer 2013 on the future of the project. If approved, a full application on construction is envisioned for 2017. So far the project has been national, but recently first steps have been taken to open it for international participation. Notably, Sweden has been pioneering in this respect with the collaboration effort being part of the Swedish counterfinancing of the French participation in ESS. Some work on ASTRID might be performed within Horizon 2020.

2.2 Myrrha

The MYRRHA project started in 1998 by SCK-CEN in collaboration with Ion Beam Applications (IBA, Louvain-la-Neuve), as an upgrade of the ADONIS project. MYRRHA is designed as a multi-purpose irradiation facility in order to support research programs on fission and fusion reactor structural materials and nuclear fuel development. Applications of these are found in Accelerator Driven Systems (ADS) and in present generation as well as in next generation critical reactors. The first objective of MYRRHA however, will be to demonstrate on one hand the ADS concept at a reasonable power level and on the other hand the technological feasibility of transmutation of Minor Actinides (MA) and Long-Lived Fission Products (LLFP) arising from the reprocessing of radio-active waste. MYRRHA aims also to help the development of the Pb-alloys technology needed for the LFR (Lead Fast Reactor) Gen-IV concept (Van den Eynde et al. 2013b).

Myrrha is the only remaining ADS project in Europe. It has been pointed out in the SNETP Strategic Research Agenda as one of the facilities of joint European interest. There have been a number of EU projects, for instance the CDT project described below, in which Myrrha has been in focus.

SKCCEN has recently appointed a consortium for balance of plant development, consisting of Areva and Ansaldo, and financed by the Belgian government with a grant of 60 M€.

2.3 EU-studies of transmutation

The research within EURATOM is composed of three major categories:

1. Fusion research.
2. Nuclear Fission & Radiation Protection.
3. Nuclear activities within the Joint Research Centres.

In the realm of Nuclear Fission & Radiation Protection, three major thematic areas can be identified:

- Radioactive Waste Management. Geological disposal of long-lived radioactive waste and the reduction of toxicity of radioactive waste through partitioning & transmutation.
- Reactor Systems. Operational safety of existing reactor systems and the potential of future reactor systems for safer, more efficient power plants and competitive nuclear industry.
- Radiation protection. Especially risks from low doses, medical uses, emergency management etc.

In addition, key cross-cutting activities are financed:

- Support for research infrastructures.
- Retaining competences and know-how in all areas of nuclear science.

The EU Strategy in P&T for Sustainability of Nuclear Energy can be summarized as follows (Bhatnagar 2008):

- Once-through cycle does not appear to be sustainable.
- Reprocessing of the spent fuel and transmutation of Minor Actinides in dedicated devices reduces radio-toxic inventory of the disposed waste in geological repositories.
- This has significant importance in non-proliferation strategy and radiological terrorism and reduces risks in case of an inadvertent human intrusion.
- A double-strata approach with sub-critical Accelerator Driven Systems (ADS) and/or critical Fast Reactors (FR) is being considered. A decision on the choice is planned in a couple of years.
- Geological disposal of the remaining waste (separation/transmutation losses) will be required.
- P&T is essential for the sustainability of nuclear energy.
- Geological disposal is indispensable for radioactive waste management.
- Both communities should work together for the future of nuclear energy.

The EU Waste Management Strategy for sustainability of nuclear energy is detailed below:

- Separation of main heavy metals reduces the volume and thermal output and extraction of heat-bearing (Sr and Cs) components permits a reduction in the needed size of the repository except possibly in salt-media that does not need this separation.
- Transmutation can reduce the half-life of most of the waste to be disposed of to a couple of hundred years overcoming the concerns of the public related to the long-life of the waste thus aiding the geological disposal community in securing a broadly agreed political consensus of waste disposal in geological repositories.
- Additional cost, additional secondary waste, activation products and intermediate-level waste and dose to workers in the process of reprocessing and transmutation itself will contribute to defining an optimal transmutation scheme.
- P&T has an added value of training many researchers in nuclear science and contributes to retaining of competence.
- Though the maximum eventual dose to human beings from a geological repository in normal scenarios is likely to be due to fission products, some evidence is appearing that minor actinides are also mobile.
- Generation IV safe advanced nuclear reactor concepts that burn waste and produce fuel for further use are gaining increased attention as a possible future course of action.
- It is clear that reprocessing of the spent fuel for a sustainable nuclear energy and fuel cycle would be required no matter what path of transmutation is followed.
- Efforts for the advanced partitioning processes should be reinforced towards pilot and test facilities for optimised separation processes in close cooperation with fuel fabrication teams and geological disposal (GD) community.
- The GD community should embrace the opportunity offered by P&T to ease the acceptance of geological repositories by society and make the best use of accepted sites by removal of long-lived and heat producing radio-nuclides from the waste.
- The GD community should especially take into account the requirements and accommodate the waste streams emanating from the advanced (minor-actinide) reprocessing systems with a view to transmutation whether in sub-critical or critical devices.

SNE-TP

The Sustainable Nuclear Energy Technology platform (SNE-TP) was officially launched on the 21st September 2007, in the presence of EU Commissioners for Science and Research, J. Potočnik, and Energy, A. Piebalgs. SNE-TP is a framework to unite all stakeholders (public-private partnership) around a common vision of sustainable nuclear energy technology research, and an attempt to mobilise a critical mass of research and innovation effort. At present, SNETP has around 110 member organizations.

In layman's terms, Technology Platforms (TP) are a type of accredited EU lobbying organizations. If a large fraction of the stakeholders – private companies, governmental organizations, universities, non-governmental organizations – within a certain technology area can create a joint platform, this TP can get recognition from EC as a discussion partner. This means that the TP gets some administrative aid from the EC, and serves the EC with advice on e.g. research funding issues and policy development.

At the launch, the SNE-TP Vision Report was presented. It highlights the role nuclear energy plays in Europe's energy mix as the main provider of low carbon electricity (providing 31 % of EU's electricity), and identifies future research, development and demonstration (RD&D) tracks that the nuclear fission sector must follow to address three objectives:

1. Maintain the safety and competitiveness of today's technologies.
2. Develop a new generation of more sustainable reactor technologies (Generation-IV Fast Reactors with closed fuel cycles).
3. Develop new applications of nuclear power such as the industrial scale production of hydrogen, desalination or other industrial process heat applications.

On 10th January 2007, the European Commission published a seminal communication, An Energy Policy for Europe (Energy Policy for Europe 2007), which for the first time underlined the benefits of nuclear energy: low carbon emissions, competitiveness, and stable prices. In the context of an anticipated increase in use of nuclear energy in the world, the Commission also recognised that "there are therefore economic benefits in maintaining and developing the technological lead of the EU in this field". This communication was endorsed by the Council in March 2007, which also committed the EU to meet ambitious objectives by 2020 of 20 % reduction in greenhouse gas emissions (compared to 1990), 20 % renewable energies in the energy mix, and 20 % reduction in energy consumption through better energy savings and management. In order to achieve these goals and realise the longer term vision of a low carbon society by 2050, the Commission identified R&D prospects of key low carbon energy technologies in a follow-up communication, the Strategic Energy Technology (SET) Plan, published on the 22nd November 2007.

Nuclear fission is cited together with other low-carbon technologies such as renewables and Carbon Capture and Storage (CCS) technology as one of the contributors to meet the 2020 challenges. By maintaining "competitiveness in fission technologies, together with long term waste management solutions", fission energy will continue to be leading low-carbon energy technology in Europe. This objective is to be achieved by acting now to "complete the preparations for the demonstration of a new generation (Gen-IV) of fission reactors for increased sustainability". From 2040 onwards, it is envisaged that this new generation of Fast Neutron Reactors will be operating in parallel to the advanced Gen-III Light Water Reactors now being built in Europe, thereby maintaining the current 1/3 share of nuclear electricity in Europe.

The SET plan has been followed up by SNETP in its more detailed Strategic Research Agenda (SRA), issued in 2009. In the SRA, sodium-cooled fast reactors (SFR) are identified as a proven technology, and should therefore be the first technology for a full-scale prototype. The design of a reactor in the 250–600 MW electric power range should be completed by the end of 2012, aiming at beginning operation in 2020. France has offered to host such a reactor, named ASTRID, and design is in progress. In parallel, research on two alternative critical fast reactor technologies, lead- and gas-cooled fast reactors, should be pursued. The aim is to select one of them in 2012 to be the second technology for deployment, and a prototype in the 50–100 MW thermal range should be in operation by 2020. Finally, the Belgian Myrrha concept is identified as the first experimental demonstrator of ADS technology.

An update of the SRA was presented in early 2013 (SNE-TP SRA 2013). This update agrees in all vital parts with the 2009 report, although it describes some parts in more detail. The only significant difference is that it has raised Myrrha somewhat in priority, and advocates parallel funding of ASTRID and Myrrha, the former as transmutation prototype reactor and the latter for materials testing with ADS demonstration and lead-cooling technology feasibility studies as important by-product.

2.3.1 GETMAT

Background

Within the GETMAT project, structural materials for the primary circuit of Generation IV reactors are investigated. This entails steels for cladding tubes, core internals, heat exchangers and SiC-SiC composites for gas cooled fast reactor cladding tubes.

A particular emphasis is given to production and characterisation of oxide dispersion strengthened steels (ODS). This is a class of steel that originally was developed in the US. The common denominator is the addition of nano-scale metal oxide particles, with the purpose of increasing the creep rupture strength. The intention is to address the poor creep rupture strength of ferritic-martensitic steels by introducing these oxide particles, thereby obtaining a creep and irradiation resistant material for high pressure applications, such as cladding tubes and heat exchangers.

Moreover, corrosion, fatigue and wear resistance of steels for lead-alloy cooled reactors are investigated. The combined effects of irradiation and corrosion in lead are studied in several irradiation tests, such as SPEED-ASTIR in BR-2, IBIS in HFR and LEXUR-II in BOR-60. The post-irradiation examination of the MEGAPIE lead-bismuth spallation target is also done within GETMAT.

In addition, modelling of radiation damage physics is carried out, in order to improve the understanding of basic mechanisms responsible for irradiation induced hardening and embrittlement of model alloys.

ODS steel manufacturing and characterisation.

The original intention was to fabricate two type of ODS-alloys with yttrium oxide dispersions. CEA managed after some difficulties to provide Fe14Cr-Y_2O_3 specimens produced by mechanical alloying. The Cr content in the French steel is determined by the desire to keep the steel cladding corrosion resistant with respect to dissolution of fuel pins in nitric acid. Test of these steels confirmed their high creep resistance. Also, friction stir welding appears to be a viable procedure for joining ODS-components without destroying the homogeneity of the oxide dispersion. The main, and rather discomforting draw-back is the large anisotropy in the micro-structure resulting from the mechanical alloying process, which means that mechanical properties are highly dependent on the direction of the applied load. At present, it seems as this would disqualify mechanically alloyed ODS-steels from application in nuclear reactors. One should note that anisotropy may be less of an issue for ODS steels fabricated using gas atomisation techniques, such as Sandvik's APMT (Jönsson et al. 2004). However, the latter approach does not give the same increase in creep strength as the mechanical alloying technique.

Corrosion & creep resistance

The GESA technique for surface alloying of steel cladding tubes with FeCrAlY has been further refined within the project. Mapping of initial Al content with respect to the final product has allowed to define the minimum concentration necessary to form a homogeneous and protective surface (Del Giacco et al. 2012a). The impact on creep rupture strength of GESA treated T91 tubes has been investigated. It was shown that whereas untreated T91 suffers severe degradation in creep resistance when tested in LBE, the GESA treated specimens performed equally well in LBE at 550 °C, as when tested in air (Weisenburger et al. 2012).

Corrosion & wear resistance

Cladding and heat exchanger tubes will be subjected to fretting from spacer grids or spacer wires. It is therefore necessary to show that fretting will not lead to loss of the protective oxide layer. An extensive set of fretting tests on T91, 15-15Ti and GESA treated T91 has been carried out in lead (Del Giacco et al. 2012b). It is shown that the depth of fretting depends significantly on temperature, load, frequency and amplitude. Low amplitudes and, somewhat surprisingly, high loads are beneficial for performance. At low temperature (450 °C), fretting assisted dissolution attack on untreated 15-15Ti may occur already after 600 hours. GESA treated material exhibit the best resistance towards fretting and provides sufficient performance for loads above 75 N and amplitudes below 15 microns.

Corrosion & irradiation

The SPEED-ASTIR experiment was carried out in BR-2. Here, T91, 316L and 15-15Ti specimens were irradiated in an LBE capsule at a temperature of 450 °C. The kinetics of oxide scale formation did not appear to deviate from that found out-of-pile. Mechanical testing showed that liquid metal embrittlement may be an issue for T91, once cracks in the protective oxide scale occur that will let LBE come in direct contact with the metallic phase. Austenitic steels appear not to be susceptible to liquid metal embrittlement, and at this temperature, corrosion is not an issue.

In LEXUR-II, irradiation in BOR-60 of specimens for mechanical testing was carried out at 350 °C in an LBE capsule and at 500 °C in a lead capsule. At the lower temperature, T91, 316L and 15-15Ti were irradiated, and at the higher T91, Fe14Cr ODS and GESA treated samples of T91 are included. As of writing, mechanical testing of samples irradiated to 8 dpa have been concluded.

The IBIS experiment in HFR featured irradiation of specimens for mechanical testing in LBE capsules maintained at 300 °C and 500 °C. Severe corrosion was observed in the capsule irradiated at higher temperature, which has been attributed to inadequate oxygen control. Even at the lower temperature, corrosion attacks were observed, which is somewhat surprising, considering the positive experience from SPEED-ASTIR and LEXUR-II. The irradiation conditions of the IBIS experiment therefore have to be carefully analysed to assess whether there has been a difference between planned and actual temperatures.

The three experiments listed above provide, together with the ongoing PIE of the MEGAPIE target, unique data on the irradiation performance of western nuclear grade steels in lead-environment. The final analysis of the results will therefore be instrumental for the qualification of these materials for MYRRHA, ALFRED and ELECTRA.

Modelling and modelling oriented experiments

The combination of modelling with modelling oriented experiments has allowed to significantly improve the understanding of irradiation induced hardening in model Fe-Cr alloys. The primary reason for hardening appears to be decoration of interstitial defect loops with Cr atoms, which then become immobile and therefore may act as obstacles for dislocation movement (Malerba et al. 2013). We may here note that Monte Carlo simulations using the two-band model constructed by Olssson and Wallenius result in exactly this type of Cr decorated loops (Olsson et al. 2005, Zhurkin et al. 2011). The existence of such loops in irradiated materials has been observed in PIE of ferritic martensitic alloys irradiated in BOR-60 (Neklyudov and Voyevodin 1994).

2.3.2 FAIRFUELS

Background

In the FAIRFUELS project, fuels and targets dedicated to transmutation of minor actinides are studied. The project includes fabrication and irradiation testing of minor actinide bearing fuels and blankets for critical fast reactors, as well as post irradiation examination of inert matrix fuels and targets previously irradiated in the EUROTRANS project. The latter activity is supported by modelling efforts.

In France, two major options are considered for transmutation of minor actinides: homogeneous and heterogeneous (Buiron et al. 2011). In the homogeneous concept, up to 4 % minor actinides are diluted in the driver fuel of a fast reactor, where the fuel is designed to achieve a plutonium breeding ratio equal to unity. The minor actinide fraction is determined by the request to burn legacy waste from light water reactors, so that an equilibrium fraction of about 1 % minor actinides in the fuel of the fast reactor can be obtained by the end of this century.

The homogeneous transmutation approach is less demanding for fuel fabrication, but has an adverse impact on safety. The addition of 4 % of minor actinides in the fuel requires to reduce the power density of the reactor by about 20 %, in order to survive the same transient over power accidents that can be managed with the equilibrium fraction of 1 % minor actinides (Zhang et al. 2010, Buiron et al. 2011).

In the heterogeneous approach, the minor actinides are concentrated in dedicated blanket elements as $(U,MA)O_2$. In these positions, the presence of minor actinides is much less degrading for reactor safety. On the other hand, the americium concentration has to be about 15 % in order for the transmutation rate in the blanket elements to exceed the production rate in the core. Since Am-241 is mainly transmuted to Cm-242, which has a half-life of 162 days, the decay heat from the alpha decay of the latter nuclide is very high. The calculations made by CEA show that after irradiating the suggested $(U,MA)O_2$ blankets until a transmutation rate for americium of 34 % is achieved, the decay heat exceeds the limit for managing the blanket assemblies in air for several thousands of days. Hence refuelling of these assemblies requires very long outages (Buiron et al. 2011).

Minor actinide bearing fuels for homogeneous transmutation

In FAIRFUELS, the so called SPHEREPAC concept for homogeneous transmutation is investigated, by fabrication and irradiation of fuel pins containing microspheres of different sizes. The flow sheet of the manufacture consists of the Sol-Gel technique to prepare microspheres of $(U,Pu)O_2$, which then are infiltrated with americium nitrate. In this way powder management can be avoided, and the amount of highly active liquid waste is minimised. No grinding of pellets is foreseen, since the fuel is compacted by vibration of the cladding tubes after filling with the spheres. The major drawback of this approach is the relatively low density achieved (85 % TD) and the concomitant low thermal conductivity.

A SPHEREPAC mixed oxide fuel pin containing 3 % americium has been fabricated by ITU, and will be irradiated in the High Flux Reactor (HFR) in 2013.

Minor actinide bearing fuels for homogeneous transmutation

The MARIOS irradiation concerns a so called ”special effects” experiment, where small $(U_{0.85},Am_{0.15})O_2$ discs are embedded in a metallic molybdenum matrix, in order to control the temperature of the blanket material and to keep it as constant as possible (D’Agata et al. 2012). In this way, the dependence of helium bubble formation and release on temperature can be studied in detail. Specimens have been fabricated by CEA, and were irradiated in HFR during 2011. Post-irradiation examination will be conducted in 2013.

Post irradiation examination

Within the EUROTRANS project (Blomgren et al. 2010), inert matrix fuels and targets were fabricated for irradiation testing in HFR (HELIOS) and Phénix (FUTURIX). Post-irradiation examination of the HELIOS specimens has been carried out in FAIRFUELS. The major outcome is that americium oxide composite CerCer targets based on MgO and zirconia matrices behave well under irradiation. CerMet composites based on molybdenum also exhibits a good behaviour at low irradiation temperature (800 °C), but may induce internal clad corrosion at higher temperature (1 100 °C).

The major drawback of the zirconia matrix is that it cannot be dissolved in nitric acid, which means that multi-recycling of such targets is excluded. MgO may cause problems during vitrification, and molybdenum tends to precipitate during the dissolution process (Ouvrier and Boussier 2012). These problems are presently being addressed in the ASGARD project by development of pre-dissolution treatments.

At NRG, post irradiation examination was made on one of the $(Pu_{0.3},Zr_{0.7})N$ fuel pins previously irradiated in the FP5 CONFIRM project. The measured burnup and swelling was in good agreement with PIE data for the other pin obtained by PSI within the EUROTRANS project. Of special value was the measurement of helium and xenon gas release, which showed a highly interesting combination of high (~80 %) release for helium, whereas less than 5 % of xenon was released (Wallenius 2011b).

The inert matrix fuels irradiated in the FUTURIX experiment are presently being extracted from the Phénix reactor, which suffered a long delay due to the failure of a crane. It is expected that the PIE will be completed before summer 2014.

Modelling

The FAIRFUELS project includes development and refinement of the thermo-mechanical simulation code MACROS, which with some success has been used to predict the outcome of the HELIOS post-irradiation examination. Certain discrepancies remain, which have to be addressed by more detailed understanding of underlying processes. In this context KTH has used a combination of *ab initio* and rate theory modelling techniques to investigate the process of helium bubble nucleation and growth in metallic molybdenum and magnesia matrices (Runevall and Sandberg 2012).

2.3.3 CDT

The design of a fast spectrum transmutation experimental facility (FASTEF), able to demonstrate the potential of transmutation and associated technology through a system working in subcritical mode (ADS) and/or critical mode, was proposed by the FP6 IP-EUROTRANS EU project. The Central Design Team project was launched to address this challenge. In reality, the work has been concentrated on design of the Myrrha concept.

During the first 18 months, a list of specifications for the design of MYRRHA/FASTEF was finalised. A reference critical core configuration was obtained after a detailed optimisation study. A categorisation and list of initiating events was obtained in the safety analysis task. Based on MYRRHA/XT-ADS design from EUROTRANS the primary system design was revised based on the new specifications. Detailed studies per component have been undertaken. A new building and plant layout has also been obtained. At the end of CDT, a revised design of MYRRHA should be delivered for the core, the primary system and the balance of plant, with the ambition to allow starting the preparation for the writing of the specifications for the different lots.

2.3.4 LEADER

Background

The LEADER project was conceived in order to carry out the design of a lead fast reactor demonstrator, showing that the technology is capable of producing electricity to the grid under commercial conditions. As such, the design should to the largest extent possible rely on existing technological solutions. The name chosen for the reactor to be developed is ALFRED, short for ”Advanced Lead Fast Reactor Demonstrator”. The intention is to build and operate ALFRED after 2025.

In addition, a 600 MWe European Lead Fast Reactor prototype (ELFR) is designed, based on the ELSY concept, including minor actinide bearing MOX fuel and spiral heat exchangers. Following successful operation of ALFRED, ELFR would be constructed in the 2030’s.

Design of ALFRED

The main design parameters of ALFRED arrived upon in the LEADER project are listed in Table 2-1. The required use of existing technology is manifested in the application of conventional MOX fuel, rather than minor actinide bearing fuel, the choice of austenitic 15-15Ti cladding instead of ferritic or ferritic-martensitic ODS steel and the use of bayonette tubes for heat exchangers, in place of the spiral heat exchanger design developed for ELSY.

Table 2-1. Major design parameters of ALFRED.

Property	Value
Thermal power	300 MW
Electrical power	130 MW
Fuel composition	$(U,Pu)O_2$
Pu fraction (inner/outer zone)	0.217/0.278
Fuel diameter (inner/outer)	2.0/9.0 mm
Fuel column height	600 mm
Cladding material	15-15Ti
Cladding diameter (inner/outer)	9.3/10.5 mm
Fuel pin height	1430 mm
Fuel pins per assembly	127
Number of fuel assemblies	171
Number of control/shutdown rods	12/4
Maximum linear power	34 kW/m
Lead coolant velocity	1.4 m/s

The peak burnup of the fuel is 100 MWd/kg for a residence time of 5 years. The relatively high Pu fraction leads to a reactivity loss of 2600 pcm per full power year, and fresh fuel will be loaded in 1/5th of the core every year to compensate for this loss of reactivity.

The major technical issue to be resolved for ALFRED is the development of a process to protect the 15-15Ti cladding tubes from corrosion at the foreseen operating temperature, which ranges from 400 °C at the core inlet to 550 °C in the hot spot of the fuel cladding. The GESA technique for surface alloying of cladding tubes that has been qualified within the GETMAT project is presently only applicable for shorter specimens, such as those foreseen for ELECTRA Moreover, a durable material for the pump impeller must be identified and validated for operation at relative velocities of 10 m/s.

All fuel elements are extend to stick out above the free lead surface level, in order to facilitate fuel handling. A tungsten ballast is used to fix the fuel elements in the core lower support-grid.

Eight heat exchangers are used to remove the heat from the primary system. A particular feature of the heat exchanger tubes is a double wall structure with a helium gap, allowing to detect any rupture of the tube. The ALFRED reactor vessel is 8.9 m tall and 8.0 m wide, corresponding to a total lead inventory of 4000 ton.

Design of ELFR

The present design concept of ELFR is characterised by the parameters listed in Table 2-2.

Table 2-2. Preliminary design parameters of ELFR.

Property	Value
Thermal power	1500 MW
Electrical power	600 MW
Fuel composition	$(U,Pu,MA)O_2$
Pu/MA fraction	0.182/0.012
Fuel diameter (inner/inner/outer)	1.0/2.0/9.0 mm
Fuel column height	1400 mm
Cladding material	15-15Ti
Cladding diameter (inner/outer)	9.3/10.5 mm
Fuel pin height	2560 mm
Fuel pins per assembly	169
Number of fuel assemblies	427
Number of control/shutdown rods	12/12
Maximum linear power	34 kW/m
Lead coolant velocity	1.5 m/s

We may note that the minor actinide fraction corresponds to the equilibrium obtained when carrying out multiple recycle of spent ELFR fuel in ELFRs. Hence, the present design features a net zero balance of minor actinides consumption/production, and is not capable of burning minor actinides from spent LWR fuel. This is a deliberate design choice, and it must be stressed that this limitation is not inherent to lead fast reactors (Tesinsky 2012).

The fuel residence time foreseen for the ELFR is 1800 days, leading to an average/peak burn-up of 52/90 MWd/kg. The total reactivity swing is therefore much smaller than for ALFRED, and hence half the core is reloaded after 900 days of irradiation.

The ELFR vessel is designed to stand 15.6 m wide and 11.0 m tall.

2.3.5 THINS

THINS (Thermal-Hydraulics of Innovative Nuclear Systems) has 24 partners with total budget over 10 MEUR and contribution from EC of 5.9 MEUR.

Thermal-hydraulics is recognized as a key scientific subject in the development of innovative reactor systems. This project is devoted to important crosscutting thermal-hydraulic issues encountered in various innovative nuclear systems, such as advanced reactor core thermal-hydraulics, single phase mixed convection and turbulence, specific multiphase flow, and code coupling and qualification. The main objectives of the project are:

- Generation of a data base for the development and validation of new models and codes describing the selected crosscutting thermal-hydraulic phenomena. This data base contains both experimental data and data from direct numerical simulations (DNS).
- Development of new physical models and modelling approaches for more accurate description of the crosscutting thermal-hydraulic phenomena such as heat transfer and flow mixing, turbulent flow modelling for a wide range of Prandtl numbers, and modelling of flows under strong influence of buoyancy.
- Improvement of the numerical engineering tools and establishment of a numerical platform for the design analysis of the innovative nuclear systems. This platform contains numerical codes of various classes of spatial scales, i.e. system analysis, sub-channel analysis and CFD codes, their coupling and the guidelines for their applications.

The project aims at achieving optimum usage of available European resources in experimental facilities, numerical tools and expertise. It seeks to will establish a new common platform of research results and research infrastructure. The main outcomes of the project should be a synergized infrastructure for thermal-hydraulic research of innovative nuclear systems in Europe. The work is going according to the plan with some minor delays. KTH participates in the project.

2.3.6 FREYA

Background

Before licensing of novel reactor concepts, it is customary to validate simulation tools by experiments conducted in zero-power facilities. E.g. did the Swedish light water program rely on KRITZ in Studsvik, where also the fast reactor R&D programme operated FR-0 during 1964–1971.

For the purpose of validating neutronic simulation tools of lead cooled fast reactors, and to develop sub-criticality monitoring systems for MYRRHA, SCK-CEN has built the so called VENUS-F facility in Mol. It consists of 30 % enriched uranium metal fuel rods surrounded by lead rods, thus simulating lead coolant. The reactor is coupled to a powerful D-T neutron generator, making sub-critical operation possible.

In its sub-critical operation mode, the facility is called GUINEVERE. The start-up of GUINEVERE on March 4 in 2010 marked the first inauguration of a new fast reactor in Europe since Super-Phenix in 1984.

Within the FREYA project, neutron flux profiles, spectral indices, control rod worth and sub-critical reactivity measurements are carried out. Sub-critical and critical configurations are studied. KTH is developing a series of training exercises for nuclear engineering students, which will be used in a pilot training course before the end of the FREYA project.

First results

An unexpected discrepancy in calculated and measured reactivity at start-up of Guinevere was only recently resolved, when it was found that lead used for the reflector of the reactor contained a significant amount of the neutron absorbing element antimony. The presence of this element was not indicated in the material certificate delivered with the purchased lead, indicating that measurements of the actual chemical composition are instrumental.

Moreover, measured flux profiles also exhibited a discrepancy with modelling in the reflector region, which could be attributed to the presence of a polyethylen component of detector, influencing the neutron spectrum to a considerable extent.

2.3.7 PELGRIMM

Background

The PELGRIMM project is dedicated to transmutation of minor actinides in oxide fuels and blankets, mainly according to the homogeneous and heterogeneous transmutation approaches studied in the

French program. These are described in more detail under the section of the FP7 FAIRFUELS project. In particular, the following activities are carried out in PELGRIMM.

Minor actinide bearing blankets

Post irradiation of the MARIOS ”special effects” experiment, where small $(U_{0.85}, Am_{0.15})O_2$ discs embedded in a metallic molybdenum matrix have been irradiated in the Dutch High Flux Reactor.

In the MARINE irradiation, full scale $(U_{0.85}, Am_{0.15})O_2$ pellets and SPHERE-PAC pins with will be irradiated in the HFR, constituting the 2nd state in the qualification programme of the minor actinide bearing blanket concept.

Minor actinide bearing driver fuels

Here, post irradiation examination of the sphere-pac $(U_{0.80},Pu_{0.17},Am_{0.03})O_2$ fuel irradiated in HFR as part of the FAIRFUELS project is made. This fuel is considered as a possible driver fuel for sodium fast reactor.

Modelling and safety assessment

Thermo-mechanical and transient analysis of pellet and sphere-pac fuels and blanket will be made with the MACROS, TRANSURANUS, GERMINAL, SIMMER and SAS4A codes. The transient analysis is made on the basis of the ESFR core design, employing sphere-pac fuel.

2.3.8 MAXIMA

The Strategic Research Agenda of the EU Sustainable Nuclear Energy Technical Platform requires new large infrastructures for its successful deployment. MYRRHA has been identified as a long term supporting research facility for all ESNII systems and as such put in the high-priority list of ESFRI. The goal of MAXSIMA (Methodology, Analysis and eXperiments for the Safety In MYRRHA Assessment) is to contribute to the safety assessment of MYRRHA.

MAXSIMA has five technical work-packages. The first contains safety analyses to support licensing of MYRRHA. Design-based, design extended and severe accident events will be studied with a focus on transients potentially leading to fuel pin failures. Fuel assembly blockage and control system failure are the least unlikely events leading to core damage. For code validation a thermal-hydraulic study of different blockage scenarios of the fuel bundle and tests of the hydrodynamic behaviour of a new buoyancy-driven control/safety system are planned.

Both are supported by numerical simulations. Safety of the Steam Generator is treated by looking at consequences and damage propagation of a SG Tube Rupture event (SGTR) and by characterising leak rates and bubble sizes from typical cracks in a SGTR.

Additionally a leak detection system and the drag on bubbles travelling through liquid LBE are studied. MOX fuel segment qualification with transient irradiations is a big step in licensing. MAXIMA include validation experiments for safety computer codes involving core damage scenarios with high temperature MOX-LBE interactions.

Fuel-coolant-clad chemistry is studied up to 1 700 °C and a core melt experiment in a reactor is prepared to assess the interaction of LBE with molten fuel.

Following the Fukushima accident, effort is put on development of enhanced passive safety systems for decay heat removal and on confinement analyses for HLM systems. A separate work package is dedicated to education and training. Beside workshops, lecture series and training sessions, virtual-safety simulator software will be developed.

2.3.9 ENSII+

The project ESNII+ “Preparing ESNII for HORIZON 2020” is presently in negotiation phase. The aim of this cross-cutting project is to develop a broad strategic approach to advanced fission systems in Europe in support of the European Sustainable Industrial Initiative (ESNII) within the SET-Plan.

To provide a technical input to the development of the roadmap several technical work packages have been included in the project.

The nuclear power safety group at KTH will possibly have a small participation in WP6: Core Safety. The goal of the work package is to support the development of the ESNII roadmap by identifying the experimental and theoretical R&D activities which are necessary for improving the present designs, as well as the existing methods, tools and databases for static and transient analysis of the ESNII critical reactor cores, i.e., the ASTRID sodium-cooled fast reactor, the ALLEGRO helium-cooled fast reactor, and the ALFRED lead-cooled fast reactor.

2.3.10 ANDES

The FP7 project Accurate Nuclear Data for nuclear Energy Sustainability, ANDES, started in May 2010 and ends April 2013. The consortium is lead by CIEMAT, Spain, and connects 20 partners, including Uppsala University. The project is supported with 3 M€ from the European Commission.

ANDES is a response to the Vision report and Strategic Research Agenda published by the European Technological Platform for a Sustainable Nuclear Energy, SNETP. These publications conclude that sustainability requires the combination of present LWR, future Advanced Fast Reactors and the waste minimization in closed cycles with Partitioning and Transmutation. To implement these new nuclear systems and their fuel cycles it is necessary to improve accuracy, uncertainties and validation of the nuclear data and models required for both those systems and the experimental and demonstration facilities involved in their validation. ANDES includes new nuclear data measurements, dedicated benchmarks based on integral experiments, and improved evaluation and modelling. These activities are specifically oriented to obtain high precision nuclear data for the major actinides present in advanced reactor fuels, reduce the uncertainties in new isotope in closed cycles with waste minimization, and better assess the uncertainties and correlations in their evaluations.

The project is divided into several work packages to address the ANDES goals. The largest work package "Measurements for advanced reactors systems (MARS)" aims at measuring total, capture, inelastic and fission cross section for key isotopes from a list of identified priorities. The work package furthermore conducts measurements of reactor kinetics and decay heat, e.g., improving the experimental information on delayed neutron emission probabilities using IGISOL in Jyväskylä, Finland.

The high energy domain, ranging from 150 to 600 MeV, is addressed in another work package. One of the aims is to improve the predictive power of models to cure identified deficiencies with impact on key parameters of ADS demonstrators. Measurements of light-ion production from 175 MeV neutrons impinging on Fe, Bi and U at TSL are also part of this package. One of the motivations is to address radioprotection problems related to tritium production in spallation targets.

Two work packages concern uncertainties and covariances of nuclear data and integral experiments for validation and constraints of uncertainties. Key objective is the adequate determination of safety and economical margins in nuclear systems. These rely directly on nuclear data uncertainties. Therefore, the proper theoretical treatment and evaluation of nuclear reactions, especially uncertainties and covariances from experimental data and nuclear model codes is addressed.

2.3.11 ERINDA

The ERINDA (European Research Infrastructure for Nuclear Data) project aims at providing a convenient platform to integrate all scientific efforts needed for high-quality nuclear data measurements in support of waste transmutation studies as well as design studies for Gen-IV systems that include an objective of producing less waste. The ERINDA consortium comprises 13 partners, equipped with nuclear data research infrastructures. The project unifies facility management, research community and stakeholders. The aim of ERINDA is to integrate all infrastructure-related aspects of nuclear data measurements and to provide access for external user to the participating facilties.

Particular emphasis are given to the following objectives:

- initiate networks leading to a stronger partnership in infrastructure management and exploitation,

- promote access and coherent use of all participating infrastructures to meet the scientific and industrial nuclear data requests, and
- merge the complementary nuclear data measurement capabilities and expertise.

To achieve its goal, different kinds of activities are carried out by the ERINDA consortium:

- Support transnational access for external users that will carry out nuclear data measurements at the ERINDA facilities.
- Support scientific visits suited to improve the success of the experimental projects carried out in the research fields of ERINDA.
- Optimise the use of the facilities for nuclear data measurements and the analysis of results by e.g. workshops.

The largest activity is on transnational access to experimental facilities. Support for such access is granted by a Program Advisory Committee, in which Jan Blomgren has served as one of its five members, being the only one with an industry affiliation. The PAC bases its selection on scientific merit, taking into account that priority should be given to user groups who have not previously used the facility, are working in countries where no such research infrastructure exists, and foster the participation of young researchers.

The proposed experiments must be situated in one of the following research domains:

- waste transmutation and minimisation, accelerator-driven systems,
- improved reactor operation and fuel management,
- advanced innovative nuclear energy systems,
- hitherto badly known properties of nuclei of special impact to the research fields of ERINDA,
- advanced methods in nuclear technologies, safety and security.

Of special interest here is that the TSL facility of Uppsala University has been one of the supported experimental facilities. Moreover, the Uppsala University research group has got support from ERINDA to build up experimental facilities at Jyväskylä, Finland for novel experiments on nuclear fission.

2.3.12 ESFR

Background

Considering that sodium is the most mature liquid metal coolant technology, it would constitute the fastest route to a fully closed fuel cycle. The relatively low boiling temperature, combined with a positive void worth, makes the sodium fast reactor relatively sensitive to perturbations, which are enhanced in the presence of americium in the fuel.

In order to address these issues, the European Sodium Fast Reactor (ESFR) project was launched. Among the major objective is to develop core design that optimises performance, safety and fuel cycle issues.

As Swedish universities do not participate in the ESFR project, this summary is based on secondary sources found in the literature.

Base design

An original design of a 3 600 MWth (1 500 MWe) SFR with conventional MOX fuel was conceived by CEA, based on previous experience from the EFR project. The relatively large void worth of this ”working horse” design was addressed by a number of design change aiming at increasing the leakage and decreasing the sodium inventory in the core. The resulting design features a lower void worth and goes under the ”SFR V2B” acronym (Sciora et al. 2009).

The average plutonium fraction in the conventional MOX fuel of the working horse design is 15.6 %, corresponding to a breeding ratio of 1.0. The fuel residence time is 2 050 equivalent full power days, leading to an average burnup of 100 GWd/ton, and a peak burnup of 145 GWd/t. The active zone sodium void worth was calculated to be +1 500 ± 100 pcm at BOL which is about 4 dollars. The void worth increases as the fuel composition reaches equilibrium, eventually reaching 2 000 ± 1 000 pcm (Sun 2012).

Reduction of void worth and mitigation of sodium boiling.

Several approaches to reducing the sodium void worth were studied in a PhD thesis carried out at PSI (Sun 2012). A solution which reduces the void worth significantly while not degrading other performance parameters appears to be the combination of sodium and neutron absorbing boron carbide layers in the top of the fuel assembly. Carrying out a transient analysis of an unprotected loss of flow, it was shown that these measures still do not allow to avoid sodium boiling in the upper part of the fuel assembly. The pressure of the sodium vapour then blocks the flow of liquid sodium inside the assembly, leading to subsequent clad melting, even at BOL. An innovative solution was proposed, consisting of opening up the fuel assembly wrapper tube in its upper part, allowing the flow of liquid sodium to pass through the interassembly gap to the upper plenum. This solution makes it possible to cool the active part of the fuel cladding tubes, even when sodium is boiling in the upper part of the assembly (Sun 2012).

Carbide core

A complementary study of an ESFR core with carbide fuel has been made (Stauff 2011). In this case, americium may not be incorporated in the driver fuel, due to the extremely poor stability of americium carbide during fabrication (Vaudez et al. 2008). Using a quasi-static approach, the performance of potential design variants under unprotected accident conditions could be assessed. Several core designs with a sodium void worth of about 4 dollars were analysed, the main conclusion being that carbide fuel SFRs are limited in their performance mainly by the swelling of the fuel, constraining the average burnup to about 65 GWd/ton.

Minor actinide transmutation

Two major options for transmutation of minor actinides in the ESFR have been studied: homogeneous and heterogeneous (Buiron et al. 2011). In the homogeneous concept, up to 4 % minor actinides are diluted in the driver fuel. The homogeneous transmutation approach is less demanding for fuel fabrication, but has an adverse impact on safety. The addition of 4 % of minor actinides in the fuel requires to reduce the power density of the reactor by about 20 %, in order to survive the same transient over power accidents that can be managed with the equilibrium fraction of 1 % minor actinides (Zhang et al. 2010, Buiron et al. 2011).

In the heterogeneous approach, the minor actinides are concentrated in dedicated blanket elements as $(U,MA)O_2$. In these positions, the presence of minor actinides is much less degrading for safety. On the other hand, the americium concentration has to be about 15 % in order for the transmutation rate in the blanket elements to exceed the production rate in the core. Since Am-241 is mainly transmuted to Cm-242, which has a half-life of 162 days, the decay heat from the alpha decay of the latter nuclide is very high. Calculations made by CEA show that after irradiating the suggested $(U,MA)O_2$ blankets until a transmutation rate for americium of 34 % is achieved, the decay heat exceeds the limit for managing the blanket assemblies in air for several thousands of days. Hence refuelling of these assemblies requires very long outages (Buiron et al. 2011).

2.3.13 ARCAS

The ARCAS project aims at comparing, on a technological and economical basis, Accelerator Driven Systems and Fast Reactors as Minor Actinide burners (Van den Eynde et al. 2013a). In order to be able to perform this comparison, one needs to have a sufficient idea on the expected minor actinide stream for a reasonably large nuclear park. Assessment of this stream is the subject of the first work package. A literature study of several previous projects focusing on fuel cycle scenario analyses has been performed together with an analysis of the current legal framework on spent fuel management and a review of current fuel reprocessing and fabrication techniques; some relevant data were obtained by some further elaboration of available simulations. This has resulted in a reference minor actinide mass flow (low and high boundaries have been estimated) and isotopic vector that can be expect based on the first PATEROS (EC-FP6) reference scenario.

In parallel, two work packages are establishing the reference Fast Reactor and Accelerator Driven System to be used in the ARCAS project. The main concern here is the maximal loading (from the point of view of safety and operations) of minor actinides in the core of either system. A brief overview of open studies has been done with a special emphasis on the EU project CP-ESFR above (3 600 MWth SFR concept), for which a set of activities have been directed towards MA transmutation. For the ADS, the EFIT design (EC-FP6 IP-EUROTRANS) has been selected as the reference case. A parametric study on relevant safety parameters (delayed neutron fraction, void effect) has been performed as a function of the minor actinide loading using the reference minor actinide vector from work package 1.

The fourth work package is responsible for a schematic design of the fuel reprocessing facilities and fuel fabrication facilities in support of the fuel cycle involving the fast reactors and accelerator driven systems. One critical component, and often overlooked in other studies, is the transportation issue: the cost and safety of the transportation of fuel with (high) minor actinide content.

Finally, the last work package has to gather all information from the other work packages in order to able to present a comparison between the two options of fast reactors or accelerator driven systems. With the number of units needed per GWe of LWR installed and the investment cost of a transmutation unit, the investment cost per GWe is determined. For selected nuclear evolution scenarios, the total investment cost needed for transmutation can be determined. Also, the total generating costs are compared, giving an answer to the question on how much the MA transmutation would add to the cost of kWh. The costs include both the investment, operational and fuel cycle. The fuel cycle costs consist of all the parts of the closed cycle, including reprocessing and fuel/target fabrication. An overview of the project has been presented recently (Van den Eynde et al. 2013a), whereas project results are awaited in the coming years.

2.4 EU-studies of partitioning

During the last decade and more several European framework programmes have been dedicated to the partitioning of nuclear waste for transmutation purposes. On the hydrochemical route they have been NEWPART (1996–1999) (Madic et al. 2000), PARTNEW (2000–2003) (Madic et al. 2002, 2004), EUROPART (2004–2007) (Hill et al. 2007) and presently ACSEPT (2008–2012) (Bourg et al. 2009). During these years we have seen a change from the almost impossible task of separating trivalent actinides from lanthanides passing too high separation (extraction) and finally finding molecules with appropriate separation factor and distribution values (Ekberg et al. 2008). Partially in parallel to these project investigations on pyrochemical partitioning processes have been performed PYROREP (PYROREP 2003). From EUROPART and on the two different strategies have been treated within the same project.

2.4.1 ACSEPT

ACSEPT (2008–2012, Bourg et al. 2009) was the latest in a row of European framework programme projects dedicated to the partitioning of nuclear waste for transmutation purposes. ACSEPT as well as the previous project EUROPART (2004–2007, Hill et al. 2007) combined both hydrochemical and pyrochemical partitioning research whilst the earlier projects have been separate (Hydrochemical: NEWPART (1996–1999) (Madic et al. 2000) and PARTNEW (2000–2003) (Madic et al. 2002, 2004), Pyrochemical: PYROREP (PYROREP 2003). During these years there has been a transition from trying to find molecules that can achieve an An(III)/Ln(III) separation towards trying to develop a feasible P&T processes, including the integration between partitioning and transmutation.

The ACSEPT project started in 2008 and ended in 2012 and was divided into four domains. In Domain 1, Hydrometallurgy, aqueous separation processes were considered, dedicated either to actinide or to grouped actinide separation. In addition, dissolution and conversion studies were undertaken. In Domain 2, Pyrometallurgy, pyrochemical separation processes were considered. Focus was put on the enhancement of the two reference types of process selected within EUROPART. Research efforts were also devoted to head-end steps, salt treatment for recycling and waste management. In Domain 3, Process and integration, all the experimental results were integrated with engineering and system studies, both in hydro- and pyrometallurgy, rendering relevant flow sheets and recommendations to prepare for future demonstrations at pilot level. In Domain 4, Teaching and education, a training and education program was implemented to share the knowledge among the partitioning community. This program included the conduction of two international workshops, the funding of two post-doctoral fellows as well as grants to students for attending selected training sessions, conferences and summer schools.

Domain 1: Hydrometallurgy

In earlier projects, such as EUROPART, the aim was to facilitate a two-step process (An(III)/Ln(III) co-extraction and selective An(III) extraction) following a PUREX or modified PUREX process. In ACSEPT, a hot-test has also been performed on this type of process (r-SANEX), however, focus has been put on trying to minimize the amount of process steps by developing the process ideas further (1c-SANEX, i-SANEX and GANEX). A simplified description of the relevant processes being developed in ACSEPT is given below.

Regular SANEX (r-SANEX) process

The feed solution of the *r*-SANEX process is the DIAMEX (The DIAMEX process co-extracts Am(III), Cm(III), Y(III), and Ln(III) from the PUREX raffinate.) product solution, i.e., Am(III), Cm(III), Y(III), and Ln(III) in approximately 0.5–1 M HNO_3). The *r*-SANEX process co-extracts Am(III) and Cm(III) from this feed solution, thus separating them from Y(III) and Ln(III).

Since this process has successfully been demonstrated in spiked (Modolo et al. 2008) and hot (Magnusson et al. 2009b) continuous lab-scale tests, it was decided (Bourg et al. 2011) that this process will not be further developed in ACSEPT.

1-cycle SANEX (1c-SANEX) process

The feed solution of the 1c-SANEX process is the PUREX raffinate. The 1c-SANEX process extracts *only* Am(III) and Cm(III) directly from this solution. If feasible, a 1-cycle SANEX process would make redundant the DIAMEX process. The development of a 1c SANEX in Germany is for example described in Wilden et al. (2010).

Innovative SANEX (i-SANEX) process

The *i*-SANEX process co-extracts Am(III), Cm(III), and Ln(III) + Y(III) from the PUREX raffinate. Am(III) and Cm(III) are then selectively back-extracted from the loaded organic phase, using selective hydrophilic complexing agents. Afterwards Ln(III) + Y(III) are back-extracted.

This process has also been demonstrated [4]. Finding complexing agents effective at high nitric acid concentration would make the process simpler without need for buffering and salting out reagents.

GANEX process

This two-cycle process extracts bulk uranium in the 1st cycle, generating a raffinate consisting of Np, Pu, Am, Cm (+ remaining U) + fission products + corrosion products in nitric acid. The 1st cycle is not a subject of ACSEPT. The 2nd cycle co-extracts all actinides from the 1st cycle raffinate. In a GANEX process developed and tested at CEA/Marcoule, the 2nd cycle comprises co-extraction of all actinides + Ln(III), a selective actinide back-extraction section, and a Ln(III) back-extraction section. The organic phase contains a mixture of a solvating extracting agent (DMDOHEMA) that extracts all actinides + Ln(III) from the highly acidic feed, plus a cation-exchanging phosphoric acid ester, keeping the Ln(III) in the organic phase during the selective An(III) back-extraction at low acidity.

All SANEX type processes are examples of hetergoenous recycling whilst the GANEX process is an example of a homogenous reprocessing process (Figure 2-1).

The GANEX process developed at Chalmers (Aneheim et al. 2011a), Sweden has been further investigated during the last years. The proces utilizes the extractants, CyMe4-BTBP (Ekberg et al. 2010) and TBP (Figure 2-2) in the diluent cyclohexanone.

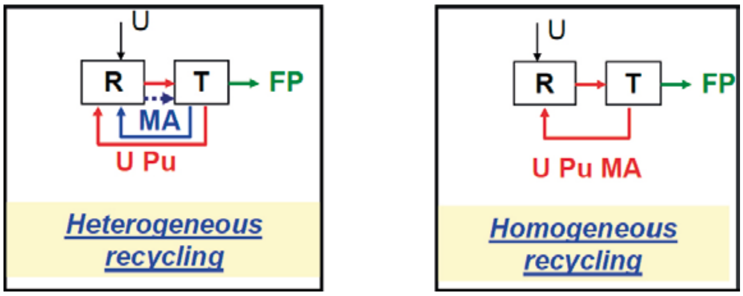

Figure 2-1. *The schematics of heterogeneous (left) and homogeneous (right) recycling.*

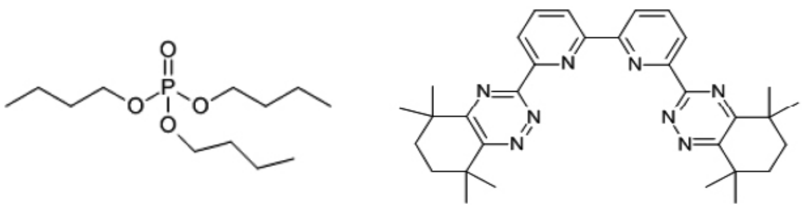

Figure 2-2. *Molecular structures of left: tri-n-butyl phosphate (TBP), right the molecular structure of (6,6'Bis(5,5,8,8-tetramethyl-5,6,7,8-tetrahydro-benzo[1,2,4]triazin-3-yl)[2,2']bipyridine) (CyMe4-BTBP).*

The solvent has proven to be resistant towards radiation and degradation (Aneheim et al. 2011a) and a problem with palladium precipitation has succesfully been treated (Aneheim et al. 2012). However, further research would focus on the replacement of cyclohexanone as diluent, or the problems related to the high water solubility of that diluent.

Several other ligands have been synthezized and screened for future applications (Lewis et al. 2011, 2012, Hudson et al. 2013), however, still the BTBP-CyMe4 seems to be the most promising one for the separation of tri-valent actinides from lanthanides.

Domain 2: Pyrometallurgy

Pyrochemical processes start with the removal of the cladding material from the fuel. This removal is followed by a dissolution of the fuel in a molten salt (except in electro refining, were no such dissolution is necessary) and a reduction of oxides to metal; separation of the wanted elements through electrowinning and finally the transfer to an immiscible liquid metal phase or a selective precipitation. The processes include a core process, but also demand a number of supplementary actions such as head-end steps, treatment of the final product in order to be compatible with following refabricating techniques, decontamination of salt and metal fluxes in order to be recyclable and treatment of waste materials.

In earlier projects (PYROREP and EUROPART) the effort was put on basic data acquisition mainly in molten chloride and on core processes assessment. Two promising core processes were developed and assessed: electro refining on solid aluminium in molten chloride and liquid-liquid reductive extraction in molten fluoride/liquid aluminium. Simultaneously, progress in the decontamination of spent chloride salt was achieved.

Following this, important technological blocks were identified as key scientific points to be studied in ACSEPT: head-end steps must be developed in fluoride and optimised in chloride. Some ancillary steps of the core process must be assessed (exhaustive electrolysis in chloride, actinide back extraction from aluminium).

Several steps towards process development have therefore been made in the latest three years. For example, the properties of Pu–Al alloys has been investigated in connection with the development of the pyrochemical methods. Electroseparation techniques in molten LiCl–KCl are being developed to group-selectively recover actinides from the mixture with fission products (Mendes et al. 2012).

The feasibility of the process has also been thermodynamically investigated (Cassayre et al. 2011).

Domain 3: Process and integration

By integrating all the experimental results within engineering and system studies, both in hydro- and pyro-metallurgy, ACSEPT has delivered relevant flow sheets and recommendations to prepare for future demonstrations at a pilot level. In hydrometallurgy, proliferation resistance and criticality assessment methodologies relevant for separation process evaluation have been identified. In addition, several molecules were synthesized in larger amounts for radiolytic or process tests.

Experiments on drops-size distributions have been performed successfully in the modified CCS device. In pyrometallurgy, a flowsheet was optimised with respect to chloride. Development of microelectrodes and LIBS technique for the on-line monitoring of molten salt processes progress very well. The reprocessing capabilities and waste management issues of CERCER and CERMET advanced fuels has been performed and used as a basis for a project within FP7 (ASGARD).

Domain 4: Teaching and education

The first international workshop took place in Lisbon, Portugal, 31 March – 2 April 2010. There were several interesting studies presented, both in an extensive poster session and orally. The proceedings from the oral presentations are available on the ACSEPT website. In addition to the possibility to present their own studies, students could also chair some selected sessions, offering not only the possibility to learn how to present data but also how to chair a meeting. The ACSEPT second international workshop took place in September, 2012 in Montpellier France. Also here interesting studies were presented and peer reviewed proceedings are available.

Student exchanges have been funded for several studies, such as the usage of lasers for achieving thermodynamic data concerning extraction systems, radiation sources for testing extraction systems ability to resist radiation damage and for the testing of extraction systems in research scale centrifuges. Students have also been funded for participating in several conferences, for example in the International Solvent Extraction Conference in Santiago, Chile, 2011 (Aneheim et al. 2011b, Löfström-Engdahl et al. 2011, Desreux et al. 2011) and the ACSEPT secondary international workshop in Montpellier etc. Also student scholarships have been distributed for participating in for example summer schools.

2.4.2 ACTINET

The objective for ACTINET has been "to provide access to external users certain parts of the facilities for training and research". This has been done through the build-up of a strong network of research facilities, standardized procedures and rules for work within these facilities. The project aimed at educating and training the members to be able to work within their areas as well as to organize a structural framework for the sustainable development of the European research area. The participating facilities have been CEA (Atalante, DPG), ITU, KIT (INE labs and beamline), FZD (IRC, ROBL) and PSI (Macro XAS beamline).

The areas of research have been divided into 4 parts. The two parts that are of interest in this report are separation science and actinide materials. In the separation science part of the program the focus has been on the separation of trivalent actinides from lanthanides. As an example of work in this area nitrogen donor ligands have been screened according to extraction and complexation behaviour. This work aimed at explaining An(III)/Ln selectivity of the ligands as well as extraction mechanisms (Panak and Geist 2013).

In the actinide materials part a fuel database has been developed at CEA. This is done in cooperation with KTH in Sweden, ITU in Germany and NRG in the Netherlands. Other areas of research are for example the production of a experimental phase diagram on the USiC system.

A list of relevant ACTINET projects is found in Table 2-3.

ACTINET has organized three summer schools with the focus on actinides. The first one in 2010 was "Analytical Innovation in the field of actinide recycling" that was held in Marcoule, France. In 2011 the school took place in Karlsruhe, Germany with focus on "Actinide Science and Applications" and the final one was in 2012 in collaboration with Plutonium futures in Cumbria, UK.

Table 2-3. ACTINET projects (JRA1 and JRA3).

D. Sedmidubsky	ICT Praque/Czech Rep.	Thermodynamic study of mixed oxide solid solution, part 1: $(Th,U)O_2$
J. O’Neill	Imp. College London	In.situ Synchrotr. Stud. of U-Ce and U-Pu oxide thin films
T. Truphemus	CEA Cadarache	EXAFS/XANES invest. of U-Pu-Oxigen Systems for mixed oxide nuclear fuel
M. Skripin	Stockholm Umi	Vibrational spectroscopic studies of hydrated Ln(III) and An(III) ions
A. Smith	Uni Cambridge	Interact. of advanced oxide fuels for Gen-IV Na cooled Fast Reactors with Na in oper. cond.
M. del H. Rojo Sanz	CIEMAT	Interaction of Nd(III)/Cm(III) and Np(IV) with gluconic acid: solubility, sorption TRLFS study…
V. Vallet	CNRS Uni Lille	Development of ab initio based force-field models for lanthanide and actinide ions
R. Thomas	CEA Cadarache	EXAFS/XANES investigations of U-Pu-Cr mixed oxides
M. Freyss	CEA	First-principles modelling of radiation damage in U nitride and U carbide
G. Law	Uni Manchester	Role of Mn in Np cycling...
T. Dekhaye	CEA Marcoule	MET study of radiation defects in self-irradiated U1-yAmyO2-x
I. Farnan	Uni Cambridge	Local probes of fundam. struct. in nucl. fuel mater.s: Pu subsitit. a. evolut. dur. irradiation…
D. Lavebtine	Uni Reading	Actinide-complexation and extraction properties of solid-supported BTP and BTPhen ligands.
S. Clark	Washington State Uni	Using ESI-MS to Understand Actinide Speciation in Separations Systems
S. Kalmikov	Lomonosov Univ	Character. of An carbide interface (UC) during the chemical dissolution in aqueous solution
R. Thomas	CEA-Cadarache	EXAFS/XANES investigations of $PuCrO_3$ saturated mixed oxide $(U,Pu)O_2$
A. Smith	UniCambridge	Interaction of advanced oxide fuels for Gen-IV sodium cooled fast reactors
A. Savero Pereira Gomes	CNRS-UniLille	Towards an accurate description of environment effects on the electronic spectra of actinides
S. Scaravaggi	Politecnico of Milano	Investigations on newly synthesized complexing agents for An selective stripping
C. Sharrad	UniManchester	Understanding molecular speciation in SANEX and GANEX separations of An from Ln
C. Sharrad	UniManchester	Understanding molecular speciation in SANEX and GANEX separations of An from Ln
V. Vallet	CNRS-UniLille	Development of ab initio based force-field modesl for lanthanide and actinide ions

Facilities

The ACTINET facilities have laboratories that allow cooperation and exchange of researchers throughout the facilities. The facilities give possibilities to handle radioactive materials at various levels of activity (hot cells, alpha-glove boxes), to work under specific conditions (inert gas, high temperature,…), to access various analytical tools (elementary and isotopic analysis, liquid, solid and interface analysis etc.), to use specific methods (laser spectroscopy, synchrotron radiation...), and to access expertise in modelling and simulation, education and means to support experimental work in actinide science.

ACTINET will be succeeded by the project TALISMAN in January 2013. One of the main activities of TALISMAN is to offer transnational access to major actinide research facilities in Europe. The number of facilities has been increased in comparison with ACTINET, the new labs being Chalmers University of Technology (Alpha and Fuel labs) in Sweden and the National Nuclear Laboratory (NNL) in Great Britain. The TALISMAN facilities are open for all scientists but the projects have to be international and at least one of the core facilities must participate.

2.4.3 ASGARD

The ASGARD project is a 4 year project funded by the EC Euroatom programme with a total budget of 9.6 M€. The project is coordinated by Chalmers University of Technology in Göteborg, Sweden. The starting date was January 2012 and therefore not so many results are available at the moment. However, it is worth noting that activities relating to dissemination, education and training started rapidly. The first part was organisation of the actinide materials session at the ATALANTE 2012 conference and recently also a winter school in fuel manufacturing co-organised with the other FP 7 projects FAIRFUELS and CINCH.

Typically, relevant research on Gen IV systems is performed in national programmes as well as in several FP7 projects. Unfortunately, today, the integration between reactor, fuel and recycling communities is lacking, in some cases resulting in discrepancies between the reactor design on one hand, and on the other hand the technological feasibility of fabrication, proper fuel behaviour, and dissolution and reprocessing of the selected fuel. The gaps filled by the ASGARD project relating to the sustainable nuclear fuel cycle is shown in Figure 2-3.

The structure of ASGARD has been optimised to reflect the differences as well as the similarities of the investigated future nuclear fuel cycles. Clearly there are separate Domains for the separate fuels, i.e. DM2 for oxides, DM3 for nitride based fuels and DM4 for carbides. A common denominator of all the fuel cycles regardless of fuel type is the conversion from the recycled actinide product to the form needed for the fuel manufacturing. In addition to this there is a significant cross cutting theme in the training and education Domain (DM1) ensuring that there is ample knowledge transfer between the different Domains and to the wider nuclear community. Thus enabling that the younger scientists educated within ASGARD will have a broad and highly qualified base for their future careers. The focus of the ASGARD project is thus on future fuels for a sustainable nuclear fuels cycle. The main problem today is to tie the recycling of the nuclear fuel to the fabrication of new fuels. As mentioned above the different reactor coolants may work differently with different fuel types and also by themselves the different fuel types have different pros and cons. The domain division of ASGARD is made to reflect this, see Figure 2-3.

Figure 2-3. ASGARD in a FP 7 EUROATOM context.

DM2, Oxides

The oxide dissolution and separation strategy is a fairly mature process being dealt with and optimised in the FP7 ACSEPT project. New separation strategies have been tested on genuine spent fuel and the selected processes will be evaluated for industrial implementation. Whereas the above is valid for actinide oxide fuels, such as MOX and/or Minor Actinide containing MOX, the dissolution and separation issues for inert matrix fuels containing ceramic MgO or metallic molybdenum (Mo), has not been investigated coherently. The ASGARD project focuses therefore mainly on the Inert Matrix Fuels (IMF) with molybdenum or magnesium-oxide, see the outline in Figure 2-5.

It is of crucial importance to take into account the behaviour of the matrix elements in the dissolution and separation processes and to check their compatibility with the future vitrification (impact on the stability of the waste and amount of generated waste).

In addition, the use of molybdenum as inert matrix poses additional challenges with respect to its redox chemistry, the need to avoid precipitation or co-precipitation, and to the necessity to recover the Mo material, which is isotopically tailored to improve the fuel behaviour.

Next to the assessment of inert matrix fuels, some basic studies will be performed to assess the dissolution of minor-actinide containing oxides, specifically with high americium and plutonium content.

A final objective of the Domain 2, which is dedicated to oxide fuels, is to address the conversion of the reprocessed solution to suitable precursors for fuel fabrication.

DM3 Nitrides

Nitride fuels constitute a high performance alternative to oxide fuel. Major advantages include a higher actinide density and a combination of high thermal conductivity with high melting temperature. The latter are particularly important in the context of transmutation in Generation IV reactors, since the addition of minor actinides to the fuel is detrimental for reactivity feedbacks. The resulting increase in fuel temperature under power transients is more easily accommodated by the larger margin to failure of the nitride fuel. The layout of Domain 3 is shown in Figure 2-6.

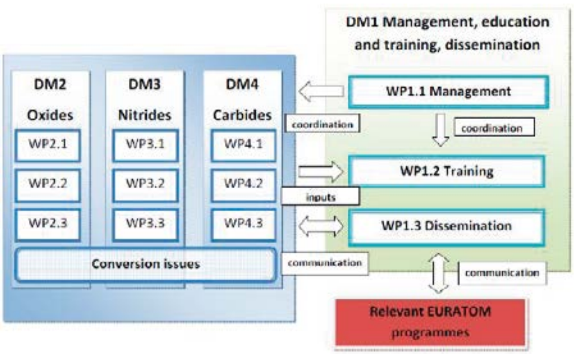

Figure 2-4. *The organisation structure of ASGARD.*

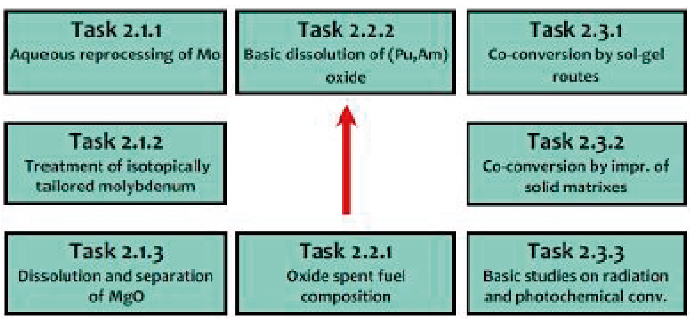

Figure 2-5. *Layout of Domain 2.*

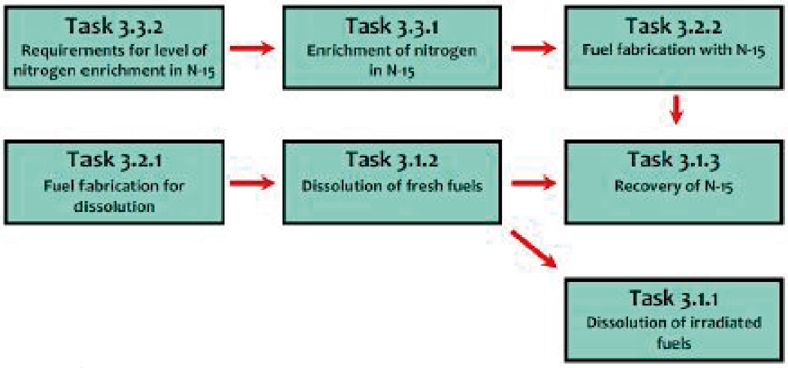

Figure 2-6. *Layout of Domain 3.*

The solubility of actinide nitrides in nitric acid is good in general. A major issue that needs to be addressed is how N-15 recycling is to be implemented. Already in the case of oxide fuels, C-14 is a major contributor to the dose commitment arising from the operation of reprocessing plants. A large scale use of nitride fuel in a closed fuel cycle would increase the C-14 production above regulatory limits if enrichment of nitrogen in N-15 is not undertaken. The cost for this enrichment might however become a significant penalty if N-15 is not recycled. Applying standard dissolution processes, dilution of N-15 in nitric acid is unavoidable. Hence a pre-treatment step involving voloxidation of the nitride fuel and recovery of the N-15 stream should be performed.

Similarly, losses of N-15 at the head end of the fabrication process must be minimized. A possible solution is the use of direct hydridation/nitridation of metallic actinides, which permits a closed gas cycle to be applied.

The ASGARD project will address all these issues, including improved processes for enrichment, the impact of fabrication route on dissolution performance and recovery of N-15.

DM4, Carbides

The high thermal conductivity of carbide fuels makes them conducive to high specific rod powers with relatively low fuel centre temperatures, the power-to-melt margin is increased and fatter (more economic) pins are facilitated. On the down side there is the potential for unacceptable fuel/clad mechanical interaction (FCMI) due to the high swelling and low plasticity of dense carbide materials. Also, the fuel fabrication process involves handling of pyrophoric powders and reprocessing is problematic because carbides dissolve in nitric acid to give a range of organic materials, some of which are flammable while others can interfere with downstream processes. In ASGARD we will mainly address the problems of fuel swelling and the issues concerning the reprocessing of carbide fuels. The layout of Domain 4 is shown in Figure 2-7.

A training and education programme will complement the main R&D programme, which aims to share the acquired knowledge among communities and generations, and maintain the nuclear expertise at the fore-front of Europe. The training and educational work package of ASGARD will work in close collaboration or will follow up other training programmes (like e.g. CINCH, ENEN and IAEA) for maximum gain and efficiency in the knowledge dissemination and strengthening the European human capital in this area.

Due to the various stages for the different fuel types, the ASGARD project will have different goals for the different fuels. However, in principle, the aim is to reach an understanding of the manufacturing and recycling of the different kind of fuels to a scientific level approaching that for oxide fuels. If, during the course of the project, decisions are taken for a selection of a specific fuel, the focus will shift towards the problems associated with that particular fuel. However, the research on the other types of fuel will continue although the effort will be reduced. The broad scope of ASGARD ensures that relevant knowledge is generated regardless of which of the fuel alternatives will be the dominating one in the future.

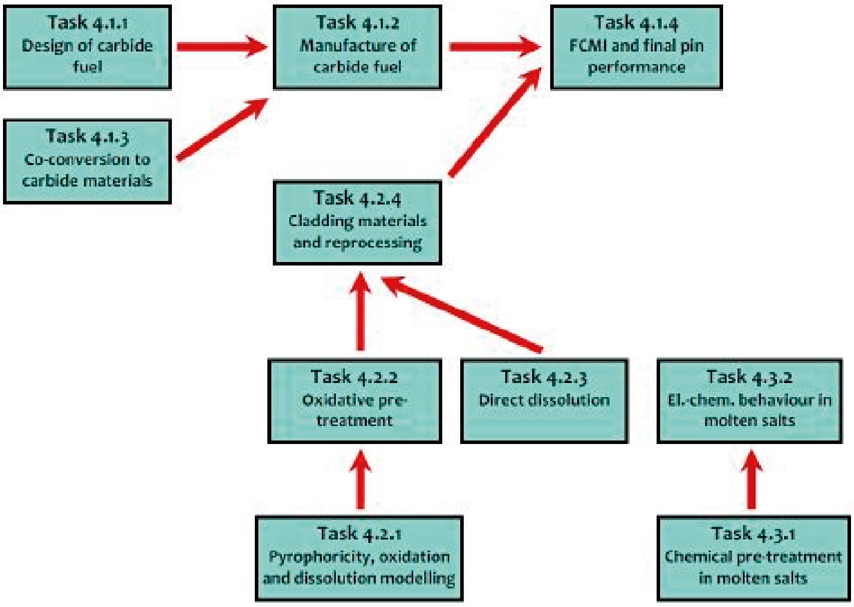

Figure 2-7. *Layout of Domain 4.*

The challenging and ambitious objectives of ASGARD will be addressed by a multi-disciplinary consortium composed of European universities, nuclear research bodies and major industrial players. Several of the partners of ASGARD are also members of the joint programme of nuclear materials organised under EERA ensuring a clear value increase by an open and simple communication line. This strong ASGARD consortium is capable to generate the fundamental knowledge for future nuclear fuel cycles and then in particular to the Gen IV system in Europe as well as to disseminate the knowledge to all relevant actors and end users over Europe. Thus the results of ASGARD will help Europe to become ready to face the challenges of the new emerging nuclear generation both in terms of people and technology.

2.4.4 CINCH

CINCH focuses on cooperation in education in nuclear chemistry in Europe. CINCH is a part of the 7th Framework Program (FP7) of the European Commission (Euratom). The project is working for the setting up of international courses and the making of an e-learning platform. The platform will offer educational tasks for several levels of students (Ph.D, life-long learning and M.Sc.). Finally, the project is working for implementing a long term sustainable strategy for nuclear chemistry education. The final workshop was held in January 2013 in Petten, the Netherlands. The workshop was part of the Winter school on Fabrication methods and Irradiation performance organized jointly by the projects FAIRFUELS and ASGARD.

2.4.5 SEARCH

In accordance with the ESNII roadmap MYRRHA will potentially be the first HLM cooled nuclear system to be deployed in Europe. The SEARCH (Safe Exploitation Related Chemistry for HLM Reactors) project aims at supporting the licensing process of MYRRHA by investigating the chemical behavior of the fuel and coolant in the reactor.

The control of the oxygen content and the management of impurities in the melt will be studied. A second critical issue in the safety assessment of a nuclear system is the compatibility of the fuel with the coolant after fuel pin leakage or a core melt. The full analyses of these scenarios using validated codes require more experimental data on basic properties of the interactions between the materials involved. For that the heat transfer coefficients of a wire-spaced fuel bundle and the basic chemical behaviour of a mixture of fuel, coolant and clad materials range will be studied at relevant temperatures.

The compatibility experiments will be done with UO_2, PuO_2 and un-radiated MOX fuel, addressing the energy release, solubility in the coolant and fuel-coolant-clad compound formation. Chalmers is responsible for this particular safety issue.

Fuel dispersion in the coolant will be simulated by a suitable numerical approach, aiming to address the migration of the fuel and the possibility to have criticality problems due to fuel accumulation. The prevention of risks to the general public will be studied by looking into the escape of radioactive materials including fission products and heavy volatile elements as Po and Hg into the environment. The kinetics and efficiency of methods to capture these elements in the cover gas system will be examined. The evaporation of Po and Hg from LBE will be measured to obtain a full data set for licensing. Issues related to Po management will be also addressed by an ab initio theoretical approach, predicting its solubility in LBE, the interaction with noble metals to select possible getters and studying formation of Po compounds.

2.4.6 EURACT-NMR

Euract-NMR is a coordination and support action of Euratom Framework 7 in order to make access possible to European nuclear licensed facilities that have recently invested in the advanced nuclear magnetic resonance spectrometers. It started in 2011 and will go on for 30 months. Three domains have been identified as research focus (cited from the web page)

- The development of actinide separation technologies through the identification of ligand selectivity, intermediates and rates of reaction by multidimensional solution state NMR spectroscopy.
- The determination of local substitution mechanisms of actinide elements into nuclear fuels and waste forms and the effects of different ordering schemes and radiation damage on physical properties.
- Determination of the magnetic and superconducting properties associated with actinide elements and actinide compounds to provide a fundamental test of the accuracy of descriptions of the 5f electron systems. Thus, underpinning ab initio modelling in nuclear applications.

Two partners in the project are KIT-INE (Karlsruhe Institute of Technology, Karlsruhe) and ITU (Institute for Transuranium Elements, Karlsruhe).

2.5 Development towards demonstration

For many years, there has been significant research on P&T worldwide, whereas few, if any, serious projects on near-industrial scale deployment have been in progress. Recently, however, France and Russia have presented plans on construction of facilities either dedicated for P&T or strongly resembling such facilities, i.e., constructed with a somewhat different focus but developing a technology of interest for future P&T.

2.5.1 France

In France, all focus is on the construction and commissioning of the sodium-cooled fast-neutron reactor ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration). ASTRID is planned to yield 600 MW electric power from a 1 500 MW thermal power. France has previously built and operated the Phénix (200 MWe) and Super-Phénix (1 200 MWe) sodium-cooled reactors, so ASTRID does not represent completely novel ground. Instead, it should be seen as re-juvenation of the earlier program that was essentially abandoned.

As described above, France has already built and operated a larger sodium-cooled fast reactor than ASTRID. The Super-Phénix project was plagued with substantial technical difficulties, and it has been argued that the development step from Phénix to Super-Phénix was too large. In that respect, ASTRID can be seen as a more moderate development step.

Another reason, however, is a slight change of focus. Super-Phénix was designed and built for power production more or less exclusively, whereas incineration of spent LWR fuel was not a major concern. For ASTRID, transmution of spent fuel and its implications on the back-end of the nuclear fuel cycle is much more prominent, although not omnipotent. This ambivalence in focus could be seen in the debate before technology freeze, in which Areva (whose interests could primarily be expected to be in power production) advocated a much larger reactor than CEA (that had a larger focus on back-end issues).

The ASTRID project has been adopted as the first step in the SNE-TP Strategic Research Agenda (SNE-TP SRA 2009). In the end of 2012, a conceptual design was released and submitted to the French Parliament for assessment. The aim is to take a decision in summer 2013 about whether to proceed. Most predictions today implicate that a positive decision will be taken, aiming at full design by 2017. An overview of the research planned for the coming years is outlined in Section 4.2.

2.5.2 Russia

General background

The former Russian ministry of atomic energy has been reorganised into the state owned company ROSATOM. Among Rosatom's subsidiaries are Rosenergoatom, responsible for all nuclear power plant operations, OKBM Gidropress, designing VVER reactors, Atomstroyexport, managing exports of nuclear build, the fuel manufacturer TVEL, four different uranium enrichment plants and the uranium mining company ARMZ.

In 2010, ROSATOM initiated the federal target program "Nuclear power of a new generation for the period 2010–2015 and up to 2020". Within this program, the "Breakthrough" project aims at demonstrating on-site recycle of spent nuclear fuel in fast neutron reactors by 2020 in order to permit commercial implementation in 2025.

Operating fast reactors and reprocessing facilities

The sodium cooled fast reactor BN-600 in Beloyarsk is continuing to produce electricity to the commercial grid. Its availability has been constant around 75 ± 5 % since 2000 (IAEA 2007b), proving the feasibility of operating sodium cooled reactors under commercial conditions. The driver fuel of the reactor is enriched UO_2, but recently test assemblies with MOX and (U,Pu)N fuel have been inserted as part of the Russian program for developing a closed nuclear fuel cycle. The current operating license of BN-600 expires in 2020.

The MAYAK reprocessing facility RT-1 is used for processing of spent fuel from VVER-440 reactors, BN-600, sub-marine and ice-breaker reactors. Since 1977, 5 000 tons of spent fuel has been reprocessed (Ivanov 2011). Considering that the nominal capacity of RT-1 is 400 tons per year, this corresponds to a utilisation factor of 35 %.

In Dimitrovgrad, the BOR-60 research reactor has obtained a life-time extension until 2014. At present, this is the world's only fast spectrum research reactor available for fuel irradiation tests under commercial conditions. The driver fuel of BOR-60 is MOX-VIPAC, a vibro-packed form which does not rely on milling or grinding of the fuel. Therefore, dust formation can be avoided. A pilot plant for dry reprocessing of spent fuel is in operation on the site, with a capacity of 1 ton per year.

Fast reactors under construction

The sodium cooled fast reactor BN-800 is under construction in Beloyarsk since 2006. It is nearing completion and is expected to become operational in 2014. Originally, it was intended to go critical on MOX fuel. Delays in the implementation of the MOX fuel fabrication plant means BN-800 most likely will start its life on enriched UOX, with a transition to MOX fuel taking place as the fuel fabrication facility comes on-line. Most of the plutonium to be used in BN-800 will come from dismantled nuclear weapons, as part of the Plutonium Management and Disposition Treaty with the United States.

In 2011, the MBIR project was approved by ROSATOM. MBIR is a 150 MW_{th} research and materials test reactor that shall substitute BOR-60. The reactor will utilise sodium coolant and MOX fuel, but shall include four different internal loops for fuel and materials tests in flowing sodium, lead, lead-bismuth and helium gas. It will therefore become the world's most versatile research tool upon its intended start of operation in 2020. Construction starts in 2013 and the total budget is estimated at 16 billion Roubles (3 billion SEK).

Fast reactors to be built

In 2012, Rosatom took the decision to build BN-1200 as a first-of-a-kind commercial sodium cooled fast reactor. The first reactor will be built on the Beloyarsk site, with intended start of operation in 2020. The capital cost for building BN-1200 is expected to be the same as for VVER-1200, meaning that the only remaining cost penalty would derive from the nuclear fuel cycle. BN-1200 will initially use MOX fuel, after which a transition to mixed nitride fuel is foreseen. Eight more units are envisaged for construction before 2030.

Rosatom has also allocated funding of 25 billion roubles (6 billion SEK) for the construction of BREST-300 in the region of TOMSK. BREST is a lead cooled fast reactor with nitride fuel, developed by the NIKIET research institute.

The technological bases for the BREST concept is the use of silicon enriched ferritic-martensitic steel EP-823 as fuel cladding. This steel is developed by IPPE and has now undergone long term (40 000 hour) tests in lead at temperatures of 600 °C. The other major technological difficulty appearing in lead cooled fast reactors is the durability of pump impellers. Here, full scale tests of lead pumps utilising a classified impeller material will be carried out in 2013–2014.

The feasibility of applying nitride fuels to improve performance and safety has been corroborated by successful irradiation of (U,Pu)N fuels with high plutonium content in BOR-60 (Zabudko et al. 2009). The observed swelling rates for these fuels was much lower than found in earlier experiments, allowing to use nitride fuel up to a burn-up of at least 12 %. At the moment, ten fuel assemblies with nitride fuel are under irradiation in BOR-60 and BN-600.

The program for qualifying nitride fuels for commercial use in fast reactors therefore has been accelerated. 17 billion roubles (4 billion SEK) has been allocated for the construction of a mixed nitride fuel factory at the Siberian Chemical Plant (SSC) complex in the Tomsk region, with the intention to be fully operational in 2017.

In addition, the SVBR-100 reactor project is progressing. This lead-bismuth cooled 100 MWe design is based on the sub-marine reactor experience of the Soviet army. Funded 50 % each by private investors and Rosatom, the AKME engineering company intends to submit a licence for build in 2014, in order to start construction of the first prototype in Dimitrovgrad in 2016. The reactor will used enriched UOX fuel for the first core, which should go critical in 2019. A progressive transition to MOX fuel and eventually mixed nitride fuel is foreseen.

Full closure of the fuel cycle

Whereas the fuel cycle facilities planned for implementation before 2020 will concern separation and recycle of plutonium, a second stage where partitioning and transmutation of minor actinides is carried out, is foreseen as an integral part of the Russian programme. In this context, the question of homogeneous or heterogeneous recycle of minor actinides has to be addressed. The particular difficulties in incorporating volatile americium in nitride fuels means that Russia is showing interest in a technology for nitride fuel fabrication presently developed in Sweden, relying on use of Spark Plasma Sintering (see Chapter on fuel fabrication).

Commercial aspects

ROSATOM claims that Generation IV technology may become commercially competitive with Generation III+ LWRs already 20 years from now. The argument for this somewhat surprising statement is that passive safety would be more easily implemented in liquid metal cooled reactors than in LWRs. Considering that it comes from an organisation involved in selling both light water and sodium technology on commercial basis, it should be seriously assessed. If holding true, nuclear industry will be on the verge of a fundamental paradigm shift much sooner than anyone may have anticipated.

2.6 Research on transmutation in various countries

2.6.1 France

Until the termination of the Super-Phénix project, France had a strong focus on sodium-cooled fast reactor, with other technologies as back-up options. At that time, however, power production was the leading motivation whereas back-end issues were of lower priority. After Super-Phénix, the research interest in France proceeded primarily along two routes. Gas-cooled fast reactors became the leading technology for research with a power production focus. In parallel, France issued a law in 1991 that two main alternatives for handling of spent nuclear fuel should be presented, whereof direct geological disposal after MOX recycling in LWR be one alternative. This triggered significant research on P&T in general, and on ADS in particular.

Over time, the focus on ADS has diminished and the interest in fast critical reactors has re-surrected. During the Chirac presidency, plans to build a reactor for incineration of spent nuclear fuel were presented. These plans were followed by an ambitious time plan, allowing no more than about 15 years for construction and commissioning. This left sodium technology as the only realistic choice.

In parallel with the research on reactors for incineration, research on partitioning has been undertaken. In this realm, France has had an even more prominent role than in transmutation. Given the presently large focus on the ASTRID project, including ancillary partitioning research and fuel development, and the large French commitment in EU-projects, a large fraction of the French research is presented in preceding sections as well as in Section 4.2, and not repeated here.

2.6.2 Russia

Like in France, the research and technical development versus P&T has changed focus over time in Russia, but for different reasons. Russia has had a strong research tradition on lead-bismuth cooled fast reactors that was originally motivated by military applications, i.e., propulsion of nuclear submarines. This technology was to quite some extent made available to civil research in the Western World during the 90's through a number of projects within ISTC and STCU[4]. This release of previously confidential technology was made at about the time when the interest in ADS was at its peak in USA and Europe, and this influx of technology made a significant boost to this research field.

Soviet Union and Russia have operated fast reactors without interruption for a long time. The focus is and has been energy production to a larger extent than in Europe, where incineration of spent fuel has been a more important aspect than in Soviet/Russia. The situation is, however, not black or white. The same technology can be used for both purposes, and to some extent they will always be intertwined. The differences on the technical side are more in the details, but the implications of the technology on society can be significantly larger than the technology differences alone might imply. If the focus is to reduce or build up the plutonium stock in a country makes a relatively modest difference in the design of the reactors and corresponding separation and fuel manufacturing techniques, but that difference has far-reaching consequences on society.

[4] International Science and Technology Centre, an initiative by USA, Canada, EU, Switzerland, Norway, Japan and South Korea to support conversion of weapons experts and technology in the former Soviet Union to civil activities. A corresponding centre operated for Ukraine, STCU (Science and Technology Centre Ukraine).

2.6.3 USA

Blue Ribbon Commission on P&T

USA has been pursuing a once-through scenario since 30 years, with the identification of Yucca Mountain as the repository site in 2002 as a pinnacle. However, in 2008 the new Obama administration announced a halt in these plans and appointed a Blue Ribbon Commission to revisit the entire situation, with the mandate to study also other scenarios. Recently, the Blue Ribbon Committee has presented its findings, summarized below (U.S. DOE 2013):

With the appropriate authorizations from Congress, the Administration currently plans to implement a program over the next 10 years that:

- Sites, designs and licenses, constructs and begins operations of a pilot interim storage facility by 2021 with an initial focus on accepting used nuclear fuel from shut-down reactor sites.
- Advances toward the siting and licensing of a larger interim storage facility to be available by 2025 that will have sufficient capacity to provide flexibility in the waste management system and allows for acceptance of enough used nuclear fuel to reduce expected government liabilities, and
- Makes demonstrable progress on the siting and characterization of repository sites to facilitate the availability of a geologic repository by 2048.

It is notable that this strategy strongly resembles the present Swedish situation, with a central interim storage (CLAB) and a licensing process for a geological repository, albeit some 30 years behind the Swedish schedule.

The prospects for P&T are commented in the report:

> The BRC concluded that "it is premature at this point for the United States to commit irreversibly to any particular fuel cycle as a matter of government policy…" and pointed out that "it is… very likely that disposal will be needed to safely manage at least some portion of the existing commercial [used nuclear fuel] inventory." Even if a closed fuel cycle were to be adopted in the future, permanent geologic disposal will still be required for residual high-level radioactive waste. Cost, non-proliferation, national security, environmental concerns, and technology limitations are some of the concerns that would need to be addressed before any future decision to close the U.S. fuel cycle through the use of recycling would be made. These factors reinforce the likelihood that the once-through fuel cycle will continue at least for the next few decades. Nevertheless, consistent with past practice and the BRC's recommendations, DOE will continue to conduct research on advanced fuel cycles to inform decisions on new technologies that may contribute to meeting the nation's future energy demands while supporting non-proliferation and used nuclear fuel and high-level radioactive waste management objectives.

Hence, there is no strategy towards implementation of P&T in a foreseeable future. Continued research is recommended. Concerning international cooperation, it is stated that:

> International cooperation has been a cornerstone of both U.S. fuel cycle R&D efforts as well as actions to reduce the global proliferation of nuclear materials. Recently, several countries, led by the U.S. and others, have come together to establish frameworks within which multi-national fuel cycle facilities could enable wider access to the benefits of nuclear power while reducing proliferation risks. The BRC recommended that the U.S. develop the capability "to accept used fuel from foreign commercial reactors, in cases where the President would choose to authorize such imports for reasons of U.S. national security." The focus of the present Strategy is on a clear path for the safe and permanent disposal of U.S. used nuclear fuel and high-level radioactive waste; however, the Administration will continue to evaluate the BRC's recommendation and will discuss with Congress the pros and cons of including it in the new waste disposal program.

It is notable that development of capacity of final storage of spent fuel and radioactive waste from other countries is recommended. Recently, Russia has signed contracts with a number of new nuclear countries on construction of light-water reactors combined with fuel supply and return of spent fuel to Russia. For obvious reasons, this full package is tempting for an acceding country, not having to develop a back-end program. It is unclear whether the recommendation to develop capacity for accepting spent fuel from other countries is intended to promote export of reactors from the US nuclear industry, or whether this is a matter of security/military aspects, or both.

Advanced Reactor Concepts Technical Review Panel

DOE sponsors a program of research, development, and demonstration related to advanced reactor concepts, both small modular reactors (SMRs) and larger systems. These advanced concepts encompass innovative reactor concepts, such as fast reactors cooled by sodium, lead, or helium; high-temperature gas-cooled reactors; and fluoride salt-cooled high-temperature reactors.

In February 2012, the U.S. Department of Energy (DOE) Office of Nuclear Energy (NE) issued a request for information (RFI) to help inform development of the DOE reactor technology research portfolio. The RFI identified eleven criteria against which the concepts would be evaluated. Reactor vendors submitted eight concept proposals in response to the RFI, and DOE-NE formed a Technical Review Panel (TRP) to evaluate the concepts, and to identify research and development (R&D) needs based on the concept submittals (U.S. DOE 2012).

The TRP was composed of nuclear reactor technology and regulation experts from national laboratories, universities, industry, and consulting firms. The individual panel members reviewed the submitted information and conducted independent checks of the applicant's self-assessment conclusions and bases. The panel members were asked to use their expert judgment to evaluate the submitted reactor concepts against the set of eleven evaluation criteria, and to identify R&D needs.

Following are the reactor concept titles that were submitted in response to the RFI:

- General Atomics – Energy Multiplier Module, (EM2) [high temperature, gas-cooled fast reactor].
- Gen4 Energy Reactor Concept [lead-bismuth fast reactor].
- Westinghouse Electric Company – Thorium-fueled Advanced Recycling Fast Reactor for Transuranics Minimization [thorium-fueled sodium-cooled fast reactor].
- Westinghouse Electric Company Thorium-fueled Reduced Moderation Boiling Water Reactor for Transuranics Minimization [thorium fueled BWR].
- Flibe Energy- Liquid Fluoride Thorium Reactor (LFTR) [thorium-fueled liquid salt reactor].
- Hybrid Power Technologies, LLC – Hybrid Nuclear Advanced Reactor Concept [gas-cooled reactor / natural gas turbine combination].
- GE-Hitachi Nuclear Energy PRISM and Advanced Recycling Center [sodium fast reactor].
- Toshiba 4S Reactor [sodium fast reactor].

The panel evaluated the advanced reactor concepts as submitted by the vendors against the eleven criteria. The criteria included:

1. Safety.
2. Security.
3. Uranium resource utilization and waste generation minimization.
4. Operational capabilities.
5. Concept maturity, operating experience, unknowns and assumptions.
6. Fuel and infrastructure considerations.
7. Assessment of market attractiveness.
8. Economics.
9. Potential regulatory licensing environment.
10. Non-proliferation.
11. Research and Development Needs.

There was no ambition to reach consensus in the TRP. Hence the report reflects various opinions and recommendations, illustrated by the following quote from the report:

> There were few differences among the concepts with regard to their evaluations for safety, security, operational capabilities and non-proliferation. Where adequate information existed,

there were notable differences with respect to fuel and infrastructure considerations, market attractiveness, economics and regulatory licensing environment. There were wide differences with regard to concept maturity and R&D needs.

The Technical Review Panel provided specific comments on each of the eleven criteria for each of the eight reactor concepts. Because of the proprietary information contained in the vendors' submittals, and reflected in the TRP member reviews, those comments have not been released in their entirety.

The advanced reactor concepts that were submitted rely on very specific technologies with relatively few truly cross-cutting needs. However, it was noted that there are some common needs that are applicable to most of the concepts. The TRP identified three need areas with issues that apply to a majority of the advanced reactor concepts:

- Development of licensing approaches for advanced reactor concepts: This will involve the development and implementation of an advanced reactor regulatory framework. It could also involve the development of advanced safety analysis tools and the development of a common verification and validation framework for these tools.
- Accelerated development of Brayton cycle technologies: This will involve efforts to accelerate the demonstration and deployment of Brayton cycle technologies. That program should focus on both the electricity producing technologies and on the coupling to the various advanced reactor technologies. The supercritical CO_2 cycle offers compelling reductions in size and cost of the power conversion system and should be a high priority.
- Development of validated advanced reactor analysis methods: This will involve the development of advanced neutronics, thermal-hydraulics, and mechanical analysis tools, and their validation to modern standards. These tools will provide credible capabilities to design advanced concepts, and understand the design margins.

To reach this set of priorities it was necessary for the TRP Chair (Phillip Finck), DOE Lead and laboratory staff to compile their sense of the collective view of the TRP. The existence of these priorities does not imply that a consensus view was obtained from TRP members.

The technology specific R&D would be for gas-cooled fast reactors, LBE-cooled fast reactors and sodium-cooled fast reactors. Technology specific R&D for other concepts is not being supported at this time due to the long term fuel cycle development requirements that would be necessary for thorium-fuelled concepts and the lack of a compelling need to couple nuclear technology to a natural gas plant.

The Fuel Cycle Technologies Program

The Fuel Cycle Technologies (FCT) program of the Department of Energy (DOE) Office of Nuclear Energy (NE) is charged with identifying promising sustainable fuel cycles and developing strategies for effective disposition of used fuel and high-level nuclear waste, enabling policymakers to make informed decisions about these critical issues. Sustainable fuel cycles will improve uranium resource utilization, maximize energy generation while minimizing waste, improve safety, and limit proliferation risk (Fuel Cycle Technologies 2012).

To effectively accomplish its mission, FCT invested nearly $112 million in FY 2012 for parallel and complementary research and development (R&D) in five technical campaign areas that span the entire nuclear fuel cycle:

- The Fuel Cycle Options Campaign is developing management processes and tools and performing integrated fuel cycle technical assessments to provide information that can be used to objectively and transparently inform and integrate Office of Fuel Cycle Technologies activities, guiding the selection of sustainable options.
- The Advanced Fuels Campaign is developing proliferation-resistant, next-generation metallic fuels for recycling of transuranics, along with advanced accident-tolerant fuel for current light water reactors.

- The Separations, Waste Forms, and Fuel Resources Campaign contributes to both a sustainable fuel cycle and improved waste management by effectively separating transuranic elements from used nuclear fuel and seeking transformational breakthroughs in waste forms with greatly improved performance.
- The Used Fuel Disposition Campaign is enabling the technology for storage, transportation, and disposal of used nuclear fuel (UNF) and wastes generated by existing and future nuclear fuel cycles.
- The Materials Protection, Accounting, and Control Technologies Campaign is developing the technologies, monitoring tools, and analysis techniques for next-generation nuclear safeguards and security, minimizing risks of proliferation.

The program has established short-, intermediate- and long-term strategic goals to implement the respective campaign objectives. Near-term goals include addressing the Blue Ribbon Commission's technical recommendations for used fuel management, increasing the focus on nuclear fuels with enhanced accident tolerance, identifying sustainable fuel cycle options for further development, introducing ”safeguards by design” concepts onto the international market, and enabling a fundamental scientific understanding of separations and waste behaviour.

In the intermediate term, the program will conduct science-based, engineering-driven research for sustainable fuel cycle options, conduct research to support extended storage of used nuclear fuel, develop the scientific basis for determining sites and procedures for disposal of used nuclear fuel, develop advanced material control instrumentation and analytical techniques, and fabricate prototype waste forms and systems for the capture and immobilization of off-gas.

In the long term, the program will demonstrate specific fuel cycle technologies. FCT will work with industry to license a retrievable fuel storage facility, support the selection and licensing of potential used fuel disposal sites, implement a fully integrated MPACT system in a large-scale processing facility, and construct facilities to demonstrate off-gas capture and waste form technologies on an engineering scale.

Figure 2-8 illustrates how these activities span the entire nuclear fuel cycle.

Nuclear chemistry

The work on advanced reprocessing in the U.S. is still concentrated on development of the UNEX and UREX+ processes. Much work is devoted to a better understanding of the processes used for the UREX+ process streams, e.g. TRUEX, NPEX, FPEX and TALSPEAK. Molten salt technology is also rather extensively investigated. The American work on P&T is very well described in ref. INL 2012.

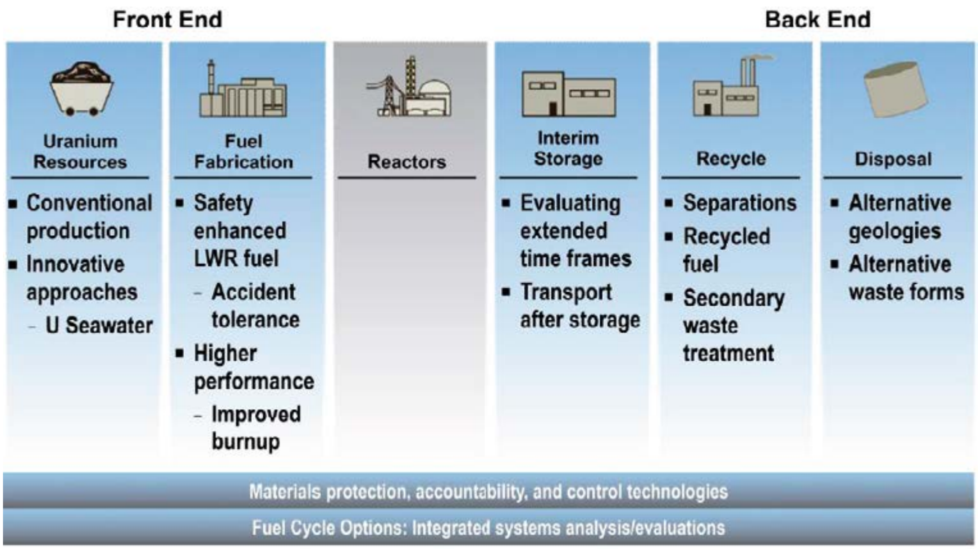

Figure 2-8. *Overview of the Fuel Cycle Technologies Program.*

The Sigma Team for Minor Actinide Separation (STMAS) was formed at the beginning of FY 2009 to enable more efficient separation methods for americium (Am) and other minor actinides (MAs) to improve the overall benefit of fuel recycle.

Strategies for Am separation are focused on either complexation or exploiting the higher oxidation states of Am. Complexation approaches seek selective binding through use of aqueous-phase complexants or specific extractants or their combinations. Manipulation of the americium oxidation state is a potentially powerful approach for a selective Am separation, but the high potentials required for oxidation of Am(III) to Am(V) and Am(VI) are challenge. Efforts are focused on methods for oxidation of Am and for separations involving Am(V) and Am(VI) separation once generated.

An Am/Eu separation factor of ~50 was demonstrated in extraction from first-cycle raffinate simulant after oxidation of Am to Am(VI) with sodium bismuthate (Mincher et al. 2011, Runde and Mincher 2011, Casely et al. 2011). Selective stripping from co-extracted Ce(IV) was successfully demonstrated. A single combined process that does the same job as the tandem TRUEX–TALSPEAK processes is advancing, with minimum actinide/lanthanide separation factors exceeding 20.

A new class of mixed-donor extractants for trivalent actinides has been designed using computational techniques. With both soft- and hard-donor groups, they should be both strong and selective (Lumetta et al. 2010a, b, 2013).

The chemistry of the TALSPEAK process is gradually being elucidated, and the understanding has led to a process variation that has much flatter pH dependence (Leggett et al. 2010, Braley et al. 2012, Grimes et al. 2010).

A new macrocyclic complexant has been shown to complex Am(III) preferentially over Cm(III), leading to a useful separation factor of 5.9, in principle sufficient for 99 % separation in six stages.

The traditional UREX+ concept of actinide (An) recovery from dissolved UNF involves an ensemble of separation steps. As such, this process manages the separation tasks, while yielding actinide product streams of high quality. Although successfully demonstrated on a laboratory scale, the feasibility of industrial implementation of a separation plant based on such a concept is in question for. The current separations philosophy of the FCR&D program sponsors viable methods of reducing the complexity of separation activities while avoiding the separation of pure plutonium. It is clear that the success of such an enterprise rests on the discovery of efficient group actinide (uranium through curium) extraction from the dissolved fuel streams.

This research effort seeks to demonstrate the feasibility of such co-extraction based on the combination of sulfur-based soft donor reagents with traditional hard-donating ligands. This separation concept relies on finding synergy between the hard donor's affinity for hexavalent and tetravalent actinides and the soft donor's preference to interact with actinides of lower charge density (Np(V), Am(III), Cm(III)). While the extraction of An(VI) and An(IV) is easily accomplished using tributyl phosphate, the lower charge density of An(III) and An(V) impedes efficient electrostatic interaction with a hard donor ligand. Thus the extraction of those tri- and pentavalent ions has to rely on their slightly softer nature, where more polarized f-electron orbitals covalently interact with soft-donor ligands. Accordingly, a combination of hard and soft donating ligands may achieve the co-extraction task.

Since the start of the project in June 2011, a methodology towards synthetic preparation of a soft-donating ligand candidate has been completed. The bifunctional sulfur ligand has been synthesized, purified and characterized. Acid degradation stability studies have been completed showing very promising results. Based on 31P nuclear magnetic resonance (NMR) results, the compound shows no signs of hydrolysis after a 60-minute contact with 1 M nitric acid. Elemental analysis is in progress (Kobayashi et al. 2010, Alyapyshev et al. 2006, Geist et al. 2006, Lumetta et al. 2002).

Research has been performed worldwide for several aqueous solvent extraction processes for Ln(III) separation from An(III). However, any large-scale separation process must be robust under high-radiation dose rates and nitric acid hydrolysis conditions. Some significant gaps in our knowledge are the effects of alpha-radiation and the kinetics/degradation products of these solvent (diluent plus extraction ligands) and metal-loaded ligand systems**.** Work is therefore done to measure the effects of gamma and alpha irradiation on solvent systems. Gamma irradiations can be performed using standard 60Co irradiators, alpha irradiations can be performed by internal isotope irradiation (244Cm, 211At), ion-beam measurements (alpha particle beams from an accelerator) and nuclear reactor-induced alpha radiolysis. The initial focus has been on the CMPO ligand in dodecane, where it has been demonstrated that the radiolytic decomposition efficiencies are approximately the same for both gamma and alpha radiolysis, and that the overall decomposition is significantly less when this formulation is irradiated in contact with acidic water (Mezyk and Mincher 2011, Swancutt 2011a, b, Elias et al. 2011).

On-Line Monitoring and U/TRU Co-deposition for Pyroprocessing are technologies that are evaluated to support development of advanced pyroprocessing methods of used nuclear fuels. Transuranics (e.g. neptunium, plutonium, americium, curium) accumulate in the electrolyte during uranium recovery operations in electrorefining. On-line monitoring provides a means of monitoring this accumulation, and uranium/transuranic (U/TRU) co-deposition provides another means of recovering the transuranics for recycle in fast reactors. The basic utilities of the two technologies were demonstrated and verified in FY 2011. Work was performed using UCl3 and PuCl3 containing electrolytes; on-line monitoring was verified by a series of voltammetry studies, and U/TRU co-deposition was verified by the recovery and analysis of gram-quantities of uranium–plutonium alloy. Results are reported e.g. in (Bryan et al. 2011a, b, c, Schroll et al. 2011, Cho et al. 2009a, b, Simpson 2011, Bezzant et al. 2011, Yoo et al. 2010, Simpson and LaBrier 2010).

During FY 2011, the radiolysis/hydrolysis test loop, located at INL, was used to study the impacts of radiolytic and hydrolytic degradation processes on the performance of the TRansUranic EXtraction (TRUEX) process (INL 2011). The successful deployment of any solvent extraction technology proposed for use in fuel cycle separations will depend upon the ability of the extractants to survive exposure to an acidic, radioactive environment. Irradiation of the ligand occurs due to the decay energy of actinides and fission products (FP) in the dissolved nuclear fuel solution. The radiation types are predominantly low linear energy transfer (LET) beta/gamma radiation from fission product decay, and high LET alpha radiation from actinide decay. The irradiation source is an MDS Nordion GammaCell 220 Excel self-contained 60Co gamma irradiator with a gamma dose rate of approximately 7.0 kGy/hr in the centre of the sample chamber. The current effective gamma dose rate in the test loop is 3.5 kGy/hr. During the solvent irradiation, the aqueous and organic phases are mixed using a centrifugal contactor (CINC V-02, USA) with the rotor replaced by a four vane mixing paddle.

The overall objective of the electrochemical separations technologies research is to improve understanding, improve the robust and cost effectiveness, develop a process monitoring capability, reduce the waste stream volume and radiological footprint, as well as create enabling technologies for the eventual commercialization and implementation of a cost effective separations technology based on electrochemical recycling technology (INL 2011). The current focal points include U/TRU co-deposition, online process monitoring for electrochemical systems, TRU and FP drawdown mechanisms, salt-metal separations, and electrochemical off-gas. These research projects are currently proof-of-concept or principle with a strong basis on the engineering-scale equipment currently operating in the Fuel Conditioning Facility (FCF) at the Materials and Fuels Complex (MFC). This close support synergy has provided a clear basis for strong extrapolation of research concepts for demonstration and validation previously in support of the objectives delineated above. The objective for the U/TRU co-deposition is to develop and demonstrate an electrochemical drawdown technology to reduce transuranics in the molten salt electrolyte. The process will recover via a co-deposition process uranium and transuranics from the salt a metallic product on a solid cathode with a corresponding anodic oxidation of used fuel.

2.6.4 Japan

Transmutation

In 2008, the Atomic Energy Commission of Japan launched the Technical Subcommittee on P&T Technology. The subcommittee launched its final report in April 2009, concluding that P&T is far from commercial deployment, and further research was recommended.

The Fukushima accident in 2011 has changed the nuclear scene dramatically. At first, political changes implicating a gradual decrease or even abandonment of nuclear power in Japan were announced. In such a scenario, ADS for P&T seemed for a while to gain some renewed interest as fast reactors were seen as impossible to pursue in a phase-out scenario. However, after the recent election, with a pro-nuclear government taking office, indications on continued utilization of nuclear power have been voiced. At present writing, it seems very difficult to predict the future of nuclear power in Japan, with a corresponding uncertainty in the future of P&T research and development.

One testimony of this volatility is provided by the Rokkasho reprocessing plant, owned by Japan Nuclear Fuel Limited and located in the village of Rokkasho-mura in northeast Aomori Prefecture. It is the successor to a smaller reprocessing plant located in Tōkai, Ibaraki prefecture. Since 1993 there has been US$ 20 billion invested in the project. At the same site there is or will also be a high level nuclear waste monitoring facility, a MOX fuel fabrication plant, a uranium enrichment plant and a low level radioactive waste landfill. After the Tōhuku earthquake and tsunami, the plan was shut down with intention never to start again. However, at present writing test operation is in progress, and the plan now is regular operation in October 2013.

Nuclear Chemistry

After the Fukushima accident, management of spent fuel in Japan got much attention and reduction of the radioactive wastes became increasingly important. P&T technology using fast reactors or accelerator-driven systems (ADS) is an important alternative discussed by the Atomic Energy Commission of Japan. To realize the transmutation of minor actinides and long-lived fission products, basic studies of various research fields are needed and therefore, equipment to obtain the experimental data using spallation target and minor actinides are required. In the framework of the J-PARC project, Japan Atomic Energy Agency (JAEA) has been promoted to construct the experimental facility for transmutation systems and performed design of Transmutation Experimental Facility (TEF). The facility consists of a Transmutation Physics Experimental Facility (TEF-P) to perform critical experiments using minor actinide bearing fuels and an ADS Target Test Facility (TEF-T) for irradiation of various structural material candidates in flowing lead-bismuth environment. According to the latest time schedule, TEF-T will be in operation within 5 years while TEF-P needs another 5 years to finish construction (Sasa et al. 2013). The TEF-T will be revised to accept proton beam up to 400MeV-133kW. A sealed-annular tube type spallation target filled by lead-bismuth eutectic alloy is considered for both low-power proton irradiation target and full stop length target for proton/neutron simultaneous irradiation. Both targets were designed to simulate the operating condition of actual lead-bismuth cooled ADS transmutor (800 MWt).

An innovative process combining the chemical dissolution of spent nitride fuel into molten chloride and the multi-stage counter current extraction of actinide elements from the molten chloride media with liquid Li-Cd alloy is reported by Satoh et al. (Satoh et al. 2013). The actinide nitrides are dissolved into molten LiCl-KCl eutectic salt as chlorides by chemical reaction with the oxidizing agents such as CdCl2. The residue materials are processed into a metal waste form. The molten salt containing actinide chlorides is transferred to the multi-stage counter current extraction of actinide elements from the molten chloride to liquid Cd using Li as a reductant. The actinides recovered in Cd are converted to nitrides by the nitridation-distillation combined method.

CRIEPI and JAEA have jointly studied the basics of actinide behaviour in molten salt and liquid metal systems, and expanded the joint study to carry out the integrated pyro-processing test and the metal fuel fabrication test for irradiation in JOYO reactor. The basic feasibility of pyrometallurgical reprocessing, such as the recovery of uranium and transuranium elements by electrorefining, has been confirmed.

Currently pyro-processing is under preliminary evaluation for treatment of the corium, mainly consisting of $(U,Zr)O_2$, formed during the accident of the Fukushima Dai-ichi nuclear power plant (Iizuka et al. 2013).

Collaboration with Japan on separations technology is carried out by the Fuel Cycle Technology Working Group (FCTWG) under the U.S. – Japan Joint Nuclear Energy Action Plan. This collaboration was relatively large during the period 2005–2010, but diminished somewhat in FY

2011. A meeting was cancelled as a consequence of the accident at the Fukushima Daiichi plant on March 11, 2011. By the late summer of 2011, the Fukushima Daiichi recovery efforts had progressed to a point that it was possible to hold a meeting of the Fuel Cycle Technology Working Group, this time in conjunction with a meeting of the U.S. – Japan Fast Reactor Working Group. Collaboration will continue i.e. on electrochemical oxide reduction process development, improved recovery efficiency in pyroprocesses, computational fluid dynamics modelling of flow in an annular centrifugal contactor, advanced aqueous extraction processes, and application of synchrotron and neutron sources to fuel cycle separation issues.

2.6.5 South Korea

Transmutation

The Korean policy on spent nuclear fuel involves a number of efforts. On the geological storage side, demonstration of a concept should be available 2016, followed by a 5-year assessment period. Final deposition is, however, not projected until around 2040.

Korea is pursuing a research program aiming at constructing a sodium-cooled fast reactor.

KALIMER-600 is a pool-type sodium-cooled reactor with a fast spectrum neutron reactor core. The core is loaded with U-TRU-Zr metal fuels. The reactor design life time is set to 60 years, which is evaluated to be long enough to provide the role imposed upon it without any degradation of the plant's efficiency. The net electricity output is 600 MWe and the net plant efficiency amounts to 39.4 %. The design should be completed by 2017, with the prospect of construction of a demonstration plant in 2028.

As an intermediate step, a significant R&D programme on direct use of spent PWR fuel in CANDU reactors (DUPIC) has been running since 1991, and today DUPIC pellets have been fabricated using PWR fuel with 65 MWd/kg burnup (Kim 2007). DUPIC fuel fabricated from lower burnup PWR pellets has been irradiated in the Hanaro research reactor. The spent Dupic fuel would then be pyrochemically processed into metallic alloy fuel for further use in sodium-cooled fast reactors.

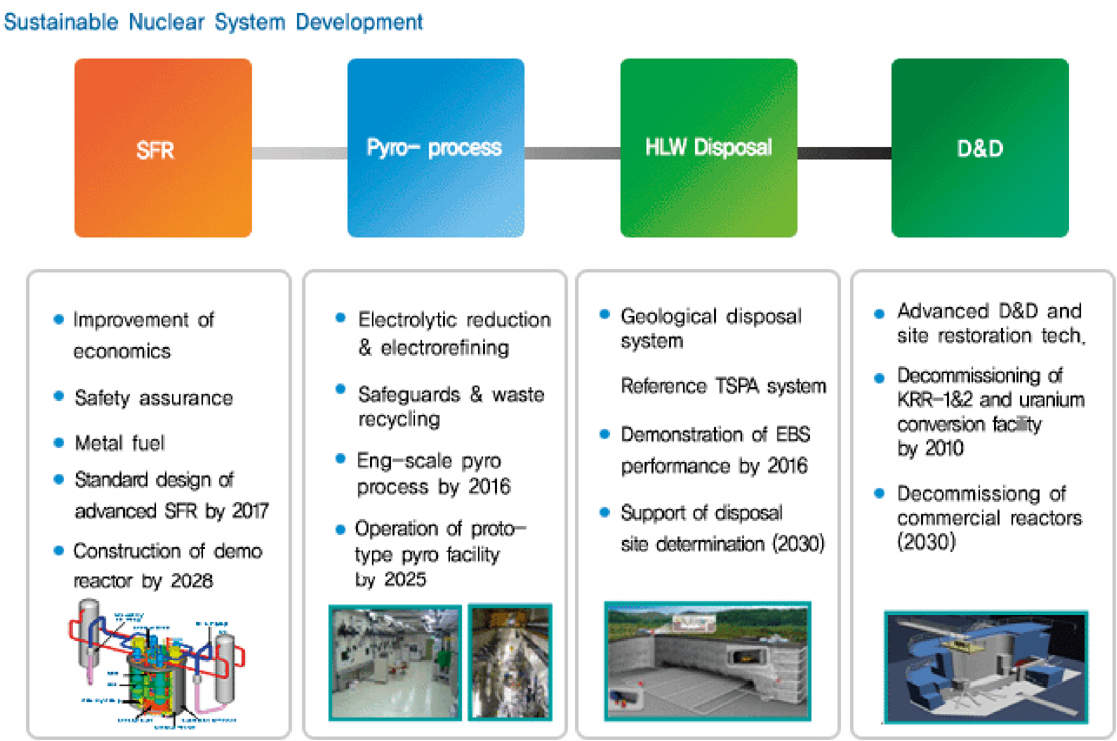

Figure 2-9. Overview of the Korean activities.

Nuclear chemistry

The issue of how to handle used nuclear fuel from nuclear power programs is important to both the United States and the Republic of Korea. Both countries have been examining options for used fuel management for some time, and this continues to be an important part of a bilateral cooperative effort. While both countries recognize that used fuel can be safely and securely maintained for decades, the need to develop cost-effective and sustainable long-term solutions, consistent with non-proliferation objectives, is also recognized. Collaboration occurs between the Idaho National Laboratory (INL), Los Alamos National Laboratory (LANL), Argonne National Laboratory (ANL), and the Korea Atomic Energy Research Institute (KAERI). The work will comprise electrochemical recycling of used LWR fuel and will expand ongoing collaborations in the development of safeguards technologies related to advanced fuel cycles and waste management. It will also include a joint evaluation of fuel cycle alternatives other than electrochemical recycling. The planned timeline for the study is 10 years, divided into three phases. All phases include joint safeguards development and evaluations of technologies important to major alternatives, such as dry cask storage. With respect to electrochemical recycling, the emphasis of the first phase is an evaluation of its laboratory-scale feasibility. In the event this evaluation is favorable, the second phase will focus on the determination of reliable integrated process operation with used LWR fuel. The third phase will evaluate the irradiation performance of fuel fabricated from recycled LWR fuel.

In support of advanced nuclear fuel cycle system development, a project supported by INL, KAERI, University of Idaho (UI), and Seoul National University (SNU) aims to design optimal electrochemical technology for recovery of zirconium from spent nuclear fuel for optimal waste management via using both modelling and experimental studies. A literature review of physical, chemical, and thermodynamic properties of zirconium (Zr) and its chloride salts in the LiCl/KCl eutectic salt system has been performed. ZrCl4 is the most stable and predominant salt species at 500 °C, limiting the available data on ZrCl2. There are no readily available published values for the diffusion coefficient of the divalent ZrCl2 molecule in the LiCl/KCl eutectic salt. The standard reduction potential and activity coefficients of Zr has been reported in literature and presented in the report.

In addition, an experiment has been designed and set up to determine parameters (e.g., diffusion coefficient, standard reduction potential, activity coefficient and exchange current density) important to the electrochemical recovery of Zr in a molten LiCl/KCl eutectic salt. The anode will be a molten cadmium pool under the salt electrolyte containing Zr metal. Current will be introduced to the anode using a tantalum lead. The cathode will also be an alumina sheathed tantalum wire. Most experiments will be run in MgO crucibles. Several runs using quartz crucibles will be selected allowing the cathodic deposition process to be digitally photographed.

2.7 IAEA

Within the framework of the project on *Technology Advances in Fast Reactors and Accelerator-driven Systems,* the IAEA has initiated a number of activities concerning the utilisation of plutonium and transmutation of long-lived radioactive waste, accelerator-driven systems, thorium fuel options, innovative nuclear reactors and fuel cycles, non-conventional nuclear energy systems, and fusion/fission hybrids.

The framework for all the IAEA activities concerning P&T is the Technical Working Group on Fast Reactors (TWG-FR). The TWG-FR acts as a catalyst for international information exchange and collaborative R&D. Given the common technical ground between plutonium utilisation R&D activities and the development of technologies for the transmutation and utilisation of long-lived fission products and actinides, both activities are performed within the framework of a single agency project: *Technology Advances in Fast Reactors and Accelerator-driven Systems*. Its present members are the following 14 IAEA member states: Belarus, Brazil, China, France, Germany, India, Italy, Japan, Kazakhstan, Republic of Korea, Russian Federation, Switzerland, United Kingdom, and United States of America, as well as the OECD/NEA, and the EU (EC). Belgium and Sweden are observers.

With regard to collaborative R&D, the IAEA has an ongoing (2002–2006) Co-ordinated Research Project (CRP) on *Studies of Advanced Reactor Technology Options for Effective Incineration* *of Radioactive Waste*, and has started a new CRP (2005–2009) on *Analytical and Experimental Benchmark Analyses of Accelerator-driven Systems (ADS)*. Recently, the TWG-FR has prepared a Technical Report on heavy liquid metal (HLM) thermal-hydraulics (IAEA 2007a).

Assessment of transmutation in fast reactors

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) launched a Joint Study that was started in 2005 and completed in 2007, but the results were not published until 2010 (IAEA 2010a). Canada, China, France, India, Japan, the Republic of Korea, the Russian Federation, and Ukraine participated. The objectives were to:

- assess the potential of a nuclear energy system based on a closed fuel cycle (CNFC) with fast reactors (FR) regarding its sustainability using the INPRO methodology,
- determine milestones for the CNFC-FR system deployment, and
- establish frameworks for, and areas of collaborative R&D work.

It was agreed to use for the assessment as a reference system a commercial CNFC–FR system, deployable in the near term (20 to 30 years) based on proven technologies, such as sodium coolant, mixed oxide (MOX) pellet fuel, and advanced aqueous reprocessing technology.

The INPRO methodology comprised an assessment in seven areas to confirm the sustainability of a nuclear energy system: economics, infrastructure, proliferation resistance, physical protection, environment (impact by stressors and availability of resources), waste management and safety. The main results could be summarized as follows:

- Availability of resources: Recycling of plutonium (together with uranium) in spent fuel of CNFC-FR systems leads to practically inexhaustible resources of fissile material (and fertile material), i.e. such a system might de facto be considered as a renewable energy source. Globally, there is sufficient spent fuel available for reprocessing Pu to be used as fuel for FR. However, in some countries with an expected high growth rate of their national nuclear power program lack of spent fuel as a resource of Pu may impede an optimal deployment of CNFC-FR systems. Thus, the Joint Study concluded that a CNFC-FR system is well suited for and might require a regional or multilateral approach as no individual country participating in the Joint Study reflects the full set of factors that favour development and deployment of such a system. Examples of important favourable factors are predicted high growth of energy demand and large resources of Pu available in spent fuel.
- Impact of stressors: CNFC-FR systems avoiding mining/enrichment steps in their fuel cycle show a significantly reduced environmental impact caused by a much lower release of non radioactive elements compared to current licensed thermal reactor systems. Additionally, the radiation dose of CNFC-FR systems on the public is demonstrated to be far below regulatory limits.
- Waste management: The CNFC–FR system meets all INPRO requirements of an effective and efficient nuclear waste management. By recycling of specific (heat producing and long lived) nuclear fission products and minor actinides in addition to plutonium (together with uranium), the CNFC–FR system has the potential to significantly reduce the heat load, mass/volume and radiotoxicity of high level waste to be deposited. The reduction of heat load enables to store more waste per volume of rock, and the removal of actinides and specific fission products from the waste decreases the time required to manage nuclear high level waste from a geological time scale (several 100 000 years) to a civilization time scale (several 100 years). However, in comparison to a once through fuel cycle (OTFC) reprocessing of spent fuel in a CNFC produces several additional secondary nuclear waste forms (e.g., losses in the processes) most of them needing geological disposal.
- Safety: Safety characteristics of the CNFC-FR system meet the current safety standards. A comparison of a CNFC-FR with a thermal reactor system showed that disadvantages of the fast neutron system were compensated by its inherent safety features and additional engineered safety measures. A probabilistic analysis performed for the Russian BN-800 fast reactor design confirmed that its innovative design features lead to a significant reduced risk of severe accidents, thus relieving the need for relocation or evacuation measures outside the plant site.

- Proliferation resistance: CNFC–FR systems show several features that result in comparable or higher proliferation resistance compared to thermal reactors with a once-through fuel cycle (OTFC). The higher proliferation resistance of CNFC-FR is justified by eliminating enrichment of uranium and avoiding the accumulation of Pu in spent fuel ("plutonium mines") in an OTFC, excluding Pu separation in advanced reprocessing technologies, and by the possibility to produce fresh fuel with a high radiation level and to reduce fuel transportation via collocation of FR and fuel cycle facilities applying pyro-processing technology. The use of a higher fissile content in the FR fuel results in a decrease of proliferation resistance in comparison to thermal reactor systems.
- Infrastructure: The INPRO basic principle in the area of infrastructure asks for the availability of regional and international arrangements to limit the necessary effort to establish the necessary infrastructure for a nuclear energy system. As stated above in the area of availability of resources, the Joint Study concluded, a CNFC–FR system is well suited for and might require such new regional or international arrangements as it is capable of converting spent fuel of all reactors into a valuable energy resource, thereby offering the opportunity expanding fuel cycle front end and backend services on a multinational basis to technology holder as well as to technology user countries. Looking at the national legal infrastructure of the countries participating in the Joint Study it was found that the legal frame work needed to operate a nuclear energy system is well established and is deemed sufficient to cover also future CNFC-FR systems. However, regional or international approaches might require new international legal infrastructure. The industrial infrastructure and human resources to design, manufacture, construct and operate a CNFC-FR system are available in most countries participating in the Joint Study.
- Economics: The designs of currently operating fast reactors with a closed fuel cycle are not completely economically competitive against thermal reactor systems or fossil power systems due to high capital costs. However, the necessary modifications of the design, such as simplifying the design, increasing the fuel burnup, constructing small series are integrated into the development programs of all Joint Study participants. These modifications will make electricity costs produced by CNFC-FR systems comparable to those of thermal reactor and fossil fuelled power plants.

Looking at the development programs of a CNFC-FR system in the countries participating in the Joint Study it was noted that:

- All countries developing fast reactor technology selected a stepwise introduction of the technology, starting with a small experimental reactor (< 50 MWth) to test the feasibility of the concept, then installing a prototype (several 100 MWth) to confirm all technical issues are resolved, thereafter constructing a first commercial size reactor (several 1 000 MWth) to proof its competitiveness, and finally install a series of commercial reactors by 2020 to 2050. A similar stepwise approach is applied for the development of the associated fuel cycle technology.
- The time schedule of instalment of the CNFC-FR system strongly depends on the global as well as on the national development of nuclear power. The higher the growth rate of nuclear power capacity either assumed globally or planned in the country, the earlier the installation of a CNFC-FR system is required to assure the availability of cheap fissile material. Differences between countries in the installation schedule are mainly caused by different predicted growth rate of the national nuclear power program.
- Countries with a large nuclear power program established for a long time and with moderate or no planned increase of their nuclear power capacity, i.e. France, Japan, or Korea, expect to accumulate enough Pu in spent fuel needed for a fast breeder program, and are therefore focusing on the development of core designs of FR with low to moderate breeding rates, e.g., Pu burners.
- Countries like China, India and Russia with rather limited contribution of nuclear power to their total energy supply but with a planned large and rapid increase of their nuclear power capacity expect not to accumulate enough Pu in spent fuel for their planned FR program, and therefore aim initially already at core designs with moderate breeding ratios and consider in the long term core designs with breeding rates as high as possible to avoid a shortage of fissile material.

The Joint Study concluded that a comprehensive program of R&D is absolutely essential in a variety of areas (especially, for economics and safety) with an inter-disciplinary approach and international collaborations wherever possible to make a CNFC-FR system a viable alternative to conventional sources of power.

As capital costs of currently operating (sodium cooled) FR were 40 % up to three times higher than capital costs of thermal reactors, several possibilities for reduction of capital costs were presented.

For the improvement of FR safety, R&D is needed to develop efficient and cost-effective shielding materials such as boride/rare earth combinations, and achieve (radiation) source reduction by adequate measures such as use of materials which do not get activated. In the Joint Study the following INPRO collaborative projects related to fast reactors have been proposed and are currently underway:

- A global architecture of nuclear energy systems based on thermal and fast reactors including a closed fuel cycle.
- Integrated approach for the design of safety grade decay heat removal system for liquid metal cooled reactor.
- Assessment of advanced and innovative nuclear fuel cycles within large scale nuclear energy system based on CNFC concept to satisfy principles of sustainability in the 21st century.
- Investigation of technological challenges related to the removal of heat by liquid metal and molten salt coolants from reactor cores operating at high temperatures.

Assessment of Partitioning

A Coordinated Research Program (CRP) on Assessment of Partitioning Processes for Transmutation of Actinides published its results in 2010 (IAEA 2010b). In the document, various aspects of partitioning processes are discussed in detail with the aim of exchanging valuable information among those involved in studying and developing viable separation methods – either within or outside the IAEA Coordinated Research Programme.

The major findings and conclusions of the report can be summarized as follows:

Technological assessment:

- Compared to aqueous processes, pyro-processes are more compact and are capable of processing spent fuels with shorter cooling times due to higher radiation resistance.
- Metal electrorefining has higher potential for recovering minor actinides, while other pyroprocesses require development of an innovative technique to recover them.
- Currently, the technical feasibility only has been established for a glass-bonded sodalite waste form using zeolite A as the waste matrix for chloride salt wastes from pyro-processes. The sodalite form, however, has relatively lower waste loading limit, compared with borosilicate glass which is suitable for high-level liquid wastes from the aqueous processes.
- While it is imperative to combine optimal options to make P&T efforts effective, the current assessment technologies for repository performance still remain at a scoping-study level, and
- Secondary wastes resulting from the various partitioning processes should also be taken into account; due consideration should be given to their treatment and conditioning.

Proliferation resistance:

- Implementation of any partitioning process supports non-proliferation of fissionable material. In planning and developing the process, special attention should be paid to neptunium and americium, and
- Mixture of actinides with low U content exhibits properties that prohibit the use for a nuclear explosive because it will emit enough neutrons by spontaneous fission that a fission chain reaction occurs immediately when a critical mass is reached.

Environmental compliance:

- The pyro-process is not well developed to minimize the process waste and to achieve a recovery of greater than 99 % especially in plant scale, while PUREX (Plutonium Uranium Extraction) process has already been demonstrated to have 99.5 % recovery, and

- By P&T deployment, the amounts of major heat emitting radio-nuclides such as 90Sr, 137Cs, 238Pu, and 241Am will be significantly reduced resulting in the simplification of both the repository and container designs.

Therefore, for the reasons outlined, it was suggested that more R&D on advanced fuel cycle approaches that will reduce the proliferation risk and improve nuclear waste management and uranium utilization, without the huge disadvantages of traditional approaches should be continued. Summarized below are areas and topics for future R&D recognized in the study:

Technological R&D:

- In-depth analysis of the nature and quantities of metal, salt and solid waste generated.
- Realising high purity inert atmosphere inside hot cells is a very big challenge in the design and operation of the plants based on dry processing.
- Need development of innovative ways for making dry processes amenable for continuous operation.
- The matrix of the waste form is an essential parameter to guarantee sustainable exclusion of the radiotoxicity from nuclides released in the environment. Therefore, efforts to fabricate a high-integrity waste form, as well as to reduce the waste generation are essential, and
- Using the already existing models, studies should be carried out to predict the waste behaviour in the proposed or adapted disposal systems.

Economics of P&T fuel cycle:

- Economics is expected to play an important role in the selection of the most viable partitioning strategy for reduction of radioactive materials, which have to be disposed of into deep geological repositories.
- Technological breakthroughs and their application might significantly enhance the economic viability of P&T, and
- Future economic assessment should take a holistic view considering all aspects of nuclear energy generation.

Finally, it was pointed out by the project team that it was not possible to provide a more detailed discussion of costs associated with the further development and implementation on a full scale of the processes discussed in the document based on their current status of development.

2.8 OECD/NEA

The nuclear development committee within the OECD/NEA has followed the development of P&T since its restart in the early 1990ies. A series of information exchange meetings have been organised (NEA 1990, 1992, 1994, 1996, 1998, 2000, 2002, 2004, 2006b, 2008, 2010, 2013). The committee has also organised a number of expert groups that have reported on various aspects of partitioning and transmutation in OECD/NEA reports. The most important of these have been summarised in previous status reports (Ahlström et al. 2004, 2007, Blomgren et al. 2010).

Under the auspices of the NEA Nuclear Science Committee (NSC), the Working Party on Scientific Issues of the Fuel Cycle (WPFC) has been established to co-ordinate scientific activities regarding various existing and advanced nuclear fuel cycles, including advanced reactor systems, associated chemistry and flow sheets, development and performance of fuels and materials, and accelerators and spallation targets. The WPFC has different subgroups to cover the wide range of scientific fields in the nuclear fuel cycle.

Created in 2002, the NEA Working Group on Lead-bismuth Eutectic (WG-LBE) technology is a WPFC subsidiary group which co-ordinates and guides LBE research in participating organizations while enhancing closer and broader-based collaboration. The aim is to develop a set of requirements and standards as well as consistent methodology for experimentation, data collection and data analyses. The results have been published in the form of a handbook (NEA 2009b). Due to a rising interest in the Pb-cooled option in the Generation IV International Forum, the WG-LBE also decided to include data and technology aspects of both LBE and Pb. The current edition of the handbook is a state-of-the-art, critical review of existing data and discrepancies, open points and perspectives for both Pb and LBE technological development.

The Expert Group on Fuel Cycle Transition Scenarios Studies was created in 2003 to consider R&D needs and relevant technology for an efficient transition from current to future advanced reactor fuel cycles. The objectives of the expert group are: i) to assemble and to organise institutional, technical and economic information critical to the understanding of the issues involved in transitioning from current fuel cycles to long-term sustainable fuel cycles or a phase-out of the nuclear enterprise; ii) to provide a framework for assessing specific national needs related to that transition.

Two reports from this group have been published previously. The report *Nuclear Fuel Cycle Transition Scenario Studies* (NEA 2009a) discusses issues related to future fuel cycles, and gives an overview of possible transition scenarios for a number of countries. In a second report, *Regional Fuel Cycle Synergies and Regional Scenarios for Europe* (Salvatores et al. 2009), Salvatores et al. proposes a regional approach for the implementation of P&T in Europe. Especially it is pointed out that with ADS in a double-strata concept, i.e., in which ADS is used primarily for MA rather than TRU incineration, the needs for a single country might not motivate an ADS facility, whereas shared regional facilities might constitute a better use of resources.

The NEA Nuclear Development Committee (NDC) has hosted an expert group carrying out a study on *Strategic and Policy Issues Raised by the Transition from Thermal to Fast Nuclear Systems* (NEA 2009c). A team of 17 experts from 11 countries and two international organizations (IAEA, EU) assessed various scenarios in deployment of fast reactors on a large scale as integral part of the nuclear energy system. Pål Efsing from Vattenfall Ringhals participated from Sweden. In short, the main findings were:

- For small countries, with a limited fleet of reactors, building the infrastructure required is unlikely to be cost effective. In those cases, regional solutions were deemed more probable.
- Countries will well-defined policies for once-through nuclear fuel cycles and advanced programs for geological direct disposal were considered less likely to adopt a fast reactor program, whereas for countries with little or no such plans fast neutron systems might offer an attractive strategy for waste management and disposal.
- Although fast reactors have been operated in a number of countries for some centuries total operation time, the technology still has not reached a maturity that allows industrial introduction without further R&D.
- Stable policies are a pre-requisite for successful introduction of a fast reactor fleet because of the long lead times required.
- Public acceptance was identified as mandatory. Non-proliferation issues need careful consideration.
- Human resource management via education and training is seen as a cornerstone.

In 2011, the Task Force on Potential Benefits and Impacts of Advanced Fuel Cycles with Partitioning and Transmutation of the WPFC presented the results of a comparative study on the impact of P&T on geological repository performance (NEA 2011). In short, the main conclusions are:

- There is no international consensus on the impact of P&T on geological storage performance. This is a consequence of the very different strategies and concepts envisioned for the geological storage.
- It is argued that P&T can reduce the heat load in a repository, allowing more efficient utilization of the repository space.
- The inventory reduction can make the uncertainty about repository performance less important, in normal operation and in particular in a disruptive scenario in which man is in contact with disposed waste. The reason for this is that the main effects are due to the hazard, i.e. radiotoxicity, rather than geology, and P&T can reduce the long-term radiotoxicity of the spent fuel.

- It is strongly emphasized that although P&T can never remove the need for geological storage, it could potentially improve the public acceptance of geologic storage.

The current NEA activities of direct relevance to P&T comprise a review of Integral Experiments for Minor Actinide Management and a study of Minor Actinide Burning in Thermal Reactors (Dujardin and Choi 2012). In addition to the above mentioned studies, the NEA also has various activities related to advanced fuel design and its impact on the fuel cycle, in particular an Expert Group on Innovative Fuels covering technical issues associated with the development of innovative fuels and cladding materials targeted for use in advanced reactors and fuel cycles, and an Expert Group on Multi-scale Modelling of Fuels in support of current fuel optimisation programmes and innovative fuel designs.

The NEA has also indicated the potential to extend its existing databases of integral experiments, especially the International Reactor Physics Benchmark Experiments (IRPhE) and International Fuel Performance Experiments (IFPE) databases, to provide the means to validate modelling methods as applied to fuels with high minor actinide content.

3 Recent results from Swedish R&D-programmes on P&T

3.1 National projects

During the last few years, two major programs have been initiated by the Swedish government where research on P&T have been significant parts. In 2009, VR was given funds for a program on GenIV research, resulting in e.g. the GENIUS project presented below. During 2010, an even larger initiative was established. France agreed to co-finance the ESS laboratory to be built in Lund, and Sweden agreed to counter-finance this contribution by giving a special grant of 11.3 M€ to the Swedish universities for collaboration with France in nuclear technology. The deal comprises funding for:

- Participation of Swedish scientists and PhD students in the design and safety analysis of the prototype reactor ASTRID, amounting to 2.8 M€.
- Access for Swedish researchers and PhD students to French laboratory facilities for joint development of fuels and materials to be tested in JHR and ASTRID, amounting to 2.0 M€.
- Access for Swedish students to French education and training reactors (ISIS and MINERVE), including costs for travel and accommodation, amounting to 2.1 M€.
- Transfer of one instrument from Uppsala University to the GANIL facility in Caen, and access for Swedish researchers to the facility for establishing the instrument and future joint work with the instrument amounting to an in kind contribution of 2.6 M€.
- Access for Swedish researchers and PhD students to the Jules Horowitz Reactor, amounting to 1.8 M€.

Part of these projects are of relevance to the present report, and are presented below.

3.1.1 GENIUS

Background

In 2009, the Swedish government allocated a fund of 50 MSEK to the Swedish research council (VR) for research on advanced nuclear energy technology. 36 MSEK of these were awarded to the GENIUS proposal from a consortium of KTH, Chalmers and Uppsala University. Janne Wallenius from KTH coordinated the project, which focused on technology for lead cooled Generation IV reactors with nitride fuel. The motivation for this choice can be found under the review of research conducted at KTH/reactor physics elsewhere in this status report. A total of ten PhD students were employed and another 15 senior scientists have been involved. The research activities were organised in three major work packages: Fuels, Materials, and Safety/Security.

Fuels

At KTH, a lab for manufacture of uranium nitride (UN) and uranium-zirconium nitride (U,Zr)N was constructed. A unique method for synthesis of pellets was elaborated, consisting of hydriding/nitriding of metallic source materials, in combination with spark-plasma sintering (SPS). Spark plasma sintering is a process whereby hot pressing of powders is assisted by a pulsed current, leading to very short process times and low residual porosity, at moderate temperature. The private company Diamorph AB operates a large SPS-machine in the premises of KTH, which has been used for spark plasma sintering of the powders produced within the GENIUS project. In Figure 3-1 the SPS equipment is shown in action.

The very fine UN powder resulting from the hydriding process, combined with the SPS process, allowed to obtain UN pellets with a density of 98 ± 0.2 % of the the theoretical maximum, which appears to be a world record. The sintering process was carried out with a holding time of three minutes at a maximum temperature of 1 650 °C. (Hollmer 2011). The appearance of the pellets is silvery greyish, as displayed in Figure 3-2. The colour reflects the highly conductive character of this ceramic, which in this aspect behaves like a metal.

Figure 3-1. Spark plasma sintering apparatus in operation. The glowing part is the graphite dye containing the powder. Image is courtesy of Diamorph AB.

Figure 3-2. Spark plasma sintered UN pellet.

Solid solution (U,Zr)N pellets may be fabricated in several ways. At KTH, hydriding/nitriding of metallic UZr alloys was carried out, where the alloy was manufactured by arc melting. Figure 3-3 shows an example of these alloys (Torres Oliver 2011).

Merja Pukari was employed as PhD student for the GENIUS project at KTH. During 2012, she spent six months in the hot laboratory of JAEA in Tokai, Japan, in order to fabricate and characterise ZrN and (Pu,Zr)N fuel pellets. The purpose of the study was to investigate the sinterability of this material as function of oxygen content. In a very careful study, it was shown that oxygen pick-up occurs both during milling and sintering processes, even when starting from highly pure materials. The fabrication route applied at JAEA consists of hydriding/nitriding of zirconium metal and carbothermic nitriding of plutonium oxide, followed by mixing, milling, pressing and conventional sintering. This is the selected reference route for manufacture of (Pu,Zr)N fuel for ELECTRA (see the chapter on ELECTRA-FCC in this status report) due to availability, safety and safeguard concerns related to the use of metallic plutonium for production of plutonium nitride.

The major outcome of the study was that the final density of sintered (Pu,Zr)N pellets may be increased by 1–2 % by oxygen pick-up and artificial addition during manufacture, the effect being larger at a higher sintering temperature. It was shown that full solubility of oxygen is possible up to at least 0.65 wt% oxygen, corresponding to 8 % light atoms. The density achieved for the (Pu,Zr)N pellets with higher oxygen content reached 88.3 % of the theoretical maximum. Two of the sintered pellets are displayed in Figure 3-4. Currently, measurements of electrical resistivity and thermal conductivity are carried out on these samples by JAEA.

At Chalmers, the main activity within the fuel work package has been to design, construct and commission a fuel fabrication lab for transuranium elements. During fall 2012, the equipment for the lab was installed, and the glove-box line was commissioned before the end of the year. The graphite sintering furnace installation is shown in Figure 3-5.

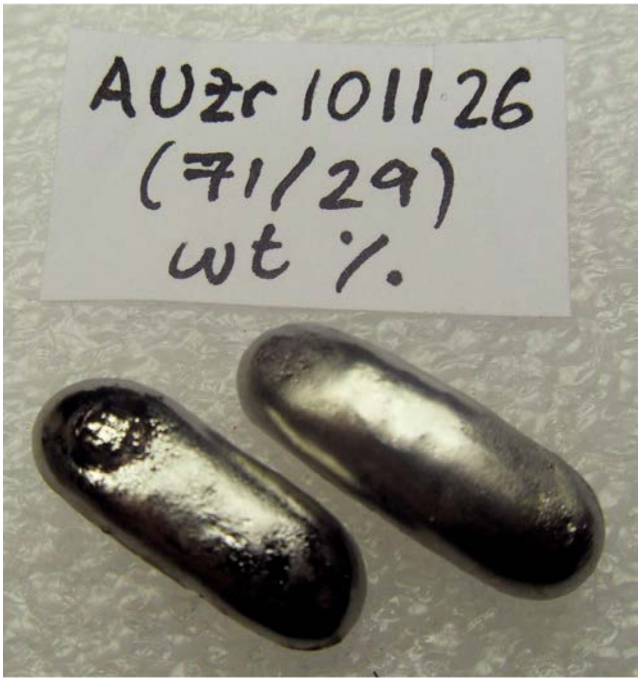

Figure 3-3. *UZr alloys used as precursor for (U,Zr)N fabrication.*

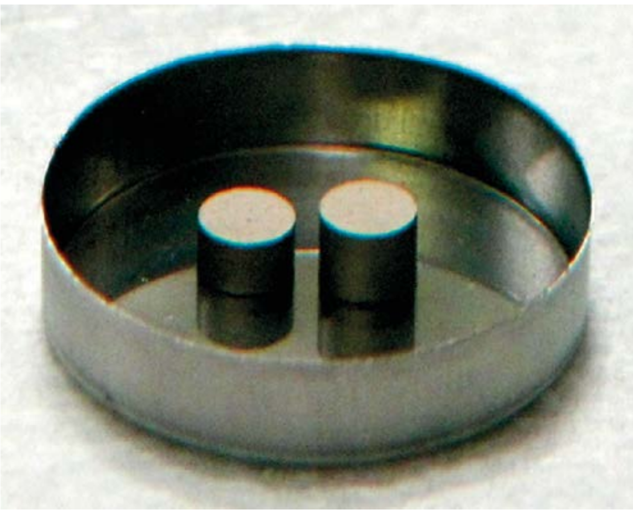

Figure 3-4. *(Pu,Zr)N pellets fabricated by KTH in JAEA labs.*

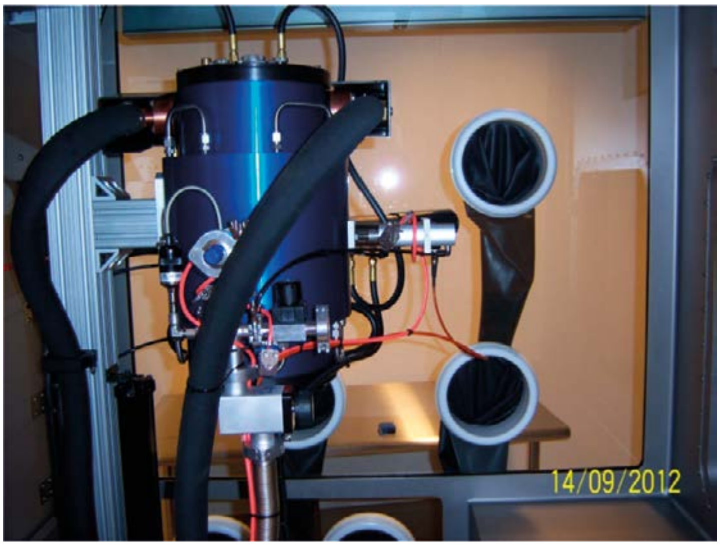

***Figure 3-5.** Graphite furnace installed in the fuel fabrication glovebox at Chalmers.*

The process intended to be used by Chalmers for manufacture of (Pu,Zr)N consists of the following steps: First, synthesis of $(Pu,Zr)O_2$ microspheres from corresponding nitrate solutions using internal gelation methods. These microspheres contain dispersed carbon, which may be added in different chemical forms, to ensure homogeneity of the dispersion. After this, carbo-thermic nitriding of the microspheres is carried out in a stream of nitrogen and hydrogen. The resulting nitride is then compacted into pellets for sintering.

A major advantage with this method is that powder handling and dust formation can be avoided. A disadvantage is that complete nitriding of the microspheres is difficult to ensure, due to inhomogeneities in the carbon dispersion.

Figure 3-6 shows a set of uranium oxide microspheres manufactured without carbon addition.

Several methods for incorporating carbon in the sol-gel process have been investigated by PhD student Marcus Hedberg, including conventional carbon black, carbon nanotubes and sucrose. Figure 3-7 shows an example of a zirconium oxide micro-sphere featuring a homogeneous carbon dispersion.

Currently, nitriding tests of zirconia microspheres are carried out, and according to plan, the first (Pu,Zr)N samples should be fabricated at Chalmers before summer 2013.

At KTH, modelling efforts are carried out to understand the behaviour of (Pu,Zr)N under irradiation. Using density functional theory (DFT) based electronic structure calculations, vacancy formation and migration activation energies have been calculated for ZrN (Pukari et al. 2010, 2013). As expected, vacancies are more likely to form on the nitrogen sub-lattice. Moreover, binding energies of helium, krypton and xenon atoms to pre-existing vacancies indicate that the fission gases will be tightly bound to such vacancies, once finding these positions. To the contrary, helium will migrate more easily. In a sub-stoichiometric nitride, such as $(Pu,Zr)N_{1-x}$, one could thus expect the ZrN matrix to function as a storage of single xenon and krypton atoms, avoiding thus the formation of fission gas bubbles that may be detrimental for swelling behaviour.

Materials

Fe-Cr steels are considered as potential structural materials for lead fast reactors, due to a combination of corrosion and radiation resistance. However, research has established that oxygen control will not be sufficient to ensure long term stability of Fe-Cr oxide scales appearing on conventional steels. Hence IPPE in Russia has developed a silicon enriched ferritic-martensitic steel (EP823), which will be used in the SVBR-100 and BREST-300 reactors to be constructed in Russia. In Germany, KIT has developed a method to surface alloy conventional steels with Fe-Cr-Al, leading to the formation of a thin and highly corrosion resistant aluminium oxide scale.

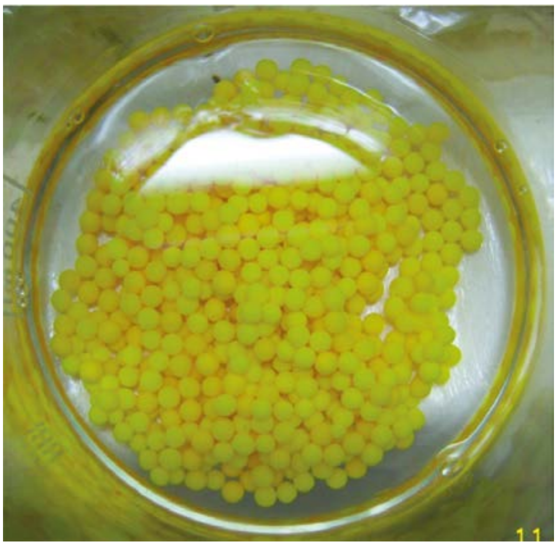

Figure 3-6. *Uranium oxide microspheres produced with the sol-gel method.*

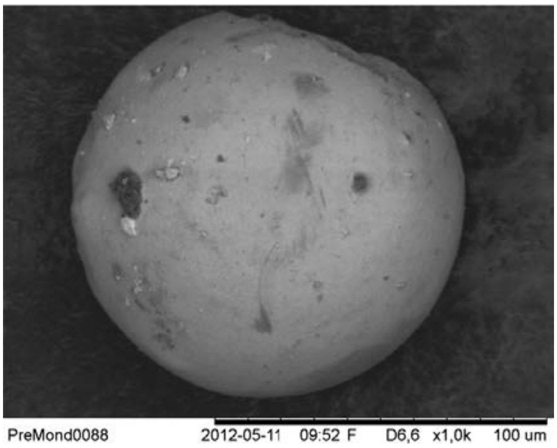

Figure 3-7. *Zirconium oxide microsphere with an homogeneous carbon dispersion.*

A furnace dedicated to long term corrosion tests of steels in liquid lead has been purchased from KIT and was installed at KTH in 2010. By now, 10 000 hour long tests of Sandvik's commercial ODS alloy APMT, as well as experimental Fe-Cr-Al and Fe-Cr-Si alloys, have been conducted by PhD student Jesper Ejenstam. These experiments are carried out at 550 °C, keeping an oxygen concentration of 10^{-7} wt% in the molten lead. The addition of aluminium or silicon leads to formation of oxide layers that prevent dissolution attack. An important feature of this layer is that it should remain thin, stable and protective over long times at elevated temperature.

In close collaboration with KTH, a set of experimental Fe-Cr-Al alloys were provided by Sandvik heating technology, that feature a reduced Cr content, as compared to their commercial Fe-20Cr-5Al ODS steel APMT. Keeping the Cr content at or below 10 %, one may avoid precipitation of Cr rich alpha-prime phase under irradiation.

Picture GENIUS.8 shows a TEM of an Fe-10Cr-6Al alloy exposed to lead at 550 °C for 10 000 hours. A remarkably thin (200 nm) alumina scale has formed on the surface, preventing corrosion attack. Presently tests are carried out on alloys with an optimised composition, aiming at further reducing the required aluminium content to a level permitting easier welding of the steel.

The aforementioned APMT and experimental Fe-Cr-Al alloys from Sandvik have been irradiated with Fe ions at the accelerator centre in Uppsala University for subsequent positron beam analysis at Chalmers by PhD student Petty Cartemo. Three different doses were achieved (0.01, 0.1 and

1.0 dpa). A highly interesting result from the preliminary analysis of Fe-10Cr-6Al is that the material contains less vacancies after irradiation than before. It is known that Fe-Al and Fe-Cr-Al alloys have a very low vacancy formation energy (Jordan and Deevi 2003). Therefore, heat treatment during manufacture, followed by quenching, may result in a supersaturation of vacancies. Irradiation of such materials may then lead to annihilation of these defects. A possible consequence might be irradiation induced softening of this material, in contrast to the hardening most often observed.

Whereas Fe-Cr alloys are considered for components such as wrapper tubes and core support grids, fuel cladding tubes for the ASTRID, MYRRHA and ELECTRA projects are foreseen to be manufactured from austenitic 15-15Ti steels. These have higher creep rupture strengths, but poorer resistance to swelling. In order to improve the understanding of austenitic structures under swelling, multi-scale modelling of radiation physics is made at KTH. PhD student Zhongwen Chang has carried out modelling of the so called dislocation bias, i.e. the tendency for dislocations to be stronger sinks for vacancies than for interstitial defects. Using a combination of atomistic calculations and the finite element methods, it is shown that elasticity theory underestimates the dislocation bias in fcc structures such as copper. This may explain previous problems of the dislocation bias model to correctly account for measured swelling rates in irradiated copper. Figure 3-9 shows the significant difference in the calculated dislocation bias factor B_d between the atomistic and the analytical calculations.

Safety

At Uppsala University, the Total Monte Carlo method (TMC) has been applied to propagate uncertainties in nuclear data measurements through construction of nuclear data libraries into neutronics simulations of safety parameters for lead cooled fast reactors. A first set of simulations for ELECTRA have been made by PhD student Erwin Alhassan for nuclear data uncertainties in Pb-208 and Pu-239. Figure 3-10 shows the obtained distribution in k-effective due to uncertainties in the nuclear data for Pu-239, using 740 random nuclear data files. The uncertainty due to nuclear data is found to be 745 ±19 pcm. It has also been shown that this uncertainty can be reduced by using benchmarks (Alhassan et al. 2014). Calculations of uncertainties in k_{eff} are a first step in calculating uncertainties in other safety parameters.

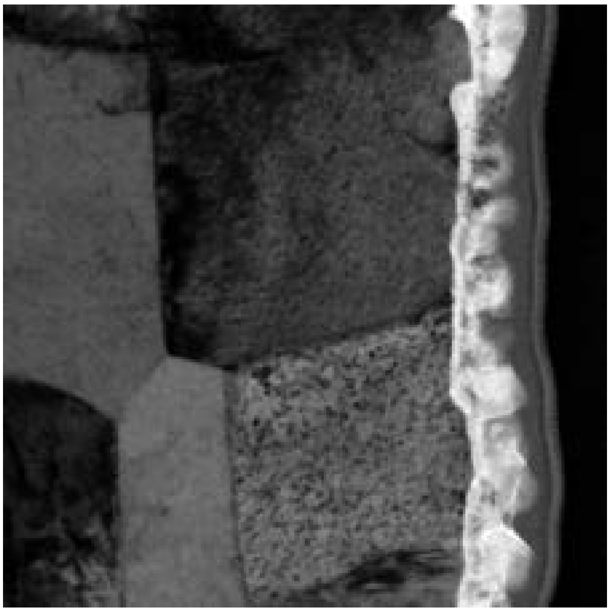

Figure 3-8. Fe-10Cr-6Al alloy exposed to lead at 550 °C for 10 000 hours. The white layer on the right side is a protective aluminium oxide.

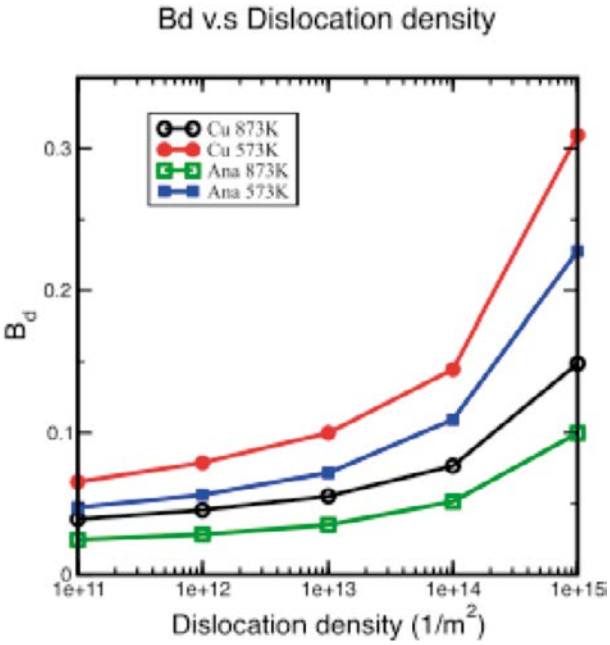

Figure 3-9. *Dislocation bias factors calculated as function of dislocation density using an atomistic approach (red and blue lines), compared to results from elasticity theory (black and green lines).*

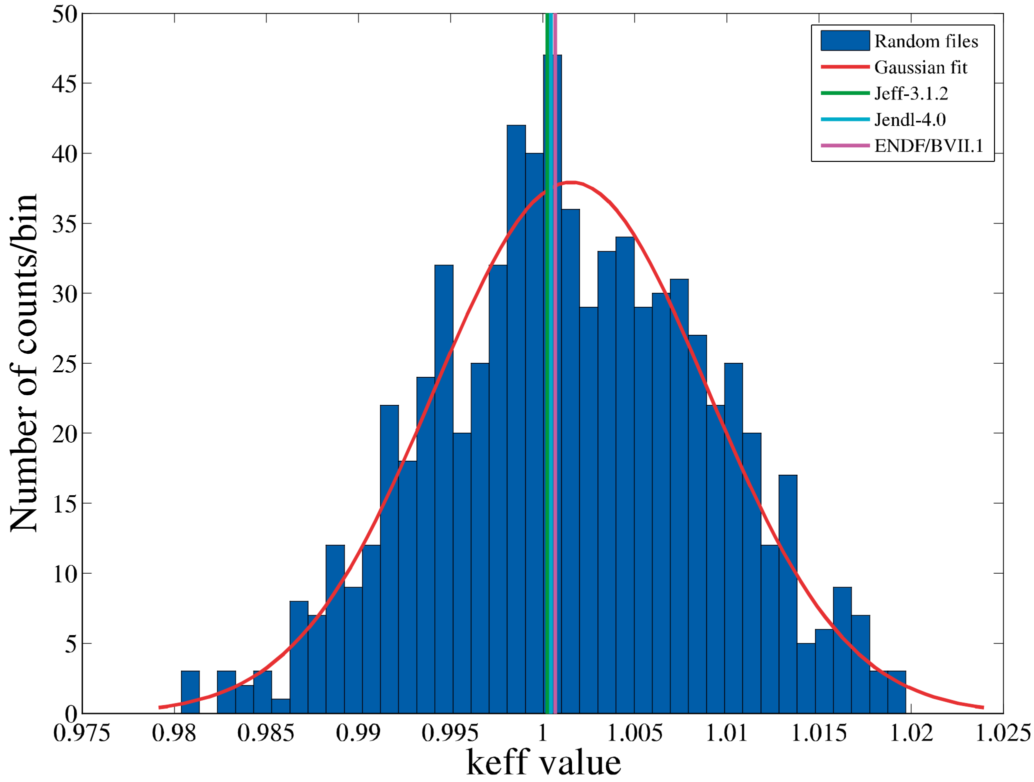

Figure 3-10. *Distribution in k_{eff} due to uncertainties in nuclear data for Pu-239. The results for the other major libraries are also shown.*

The potential for using fission chamber detectors to monitor coolant void formation in lead cooled reactors via changes in neutron spectrum has been investigated by PhD student Peter Wolniewicz in Uppsala University. Such voids can arise due to steam generator tube rupture, or fission gas release from defective fuel pins. Combining detectors sensitive to power (U-235) and spectrum (Pu-242), where the U-235 detector is located on the core boundary, and the other far away from the core it becomes possible to discriminate between spectral and power variations. Simulation of detector responses with the Monte-Carlo code Serpent showed that a signal of 2.5 % per percent void is achievable in this configuration (Wolniewicz 2012).

Conductive heat transfer between hot and cold legs in the primary circuit of liquid metal cooled reactors may cause a reduction in natural convection flow. At KTH, PhD student Roman Thiele has investigated this phenomenon using CFD simulations of ELECTRA. For this purpose, a heat exchanger design for ELECTRA was elaborated, relaying on eight sections located in the upper cold leg (see Figure 3-11). 3-D simulation of the primary system found that any by-pass flow around the core would reduce the heat removal from the core drastically. Hence, the cooling system for the rotating absorber drums must be based mainly on conduction (Thiele and Anglart 2013). Moreover, a flow reduction of as much as 25 % may result from the conductive heat transfer from hot to cold legs. The latter phenomenon must be studied in more detail.

The ultimate consequence of a severe accident where cladding tube barrier is failing will depend on the integrity of the fuel when interacting with the coolant. The compatibility of nitride fuel with lead coolant at high temperature has been investigated by KTH. Fragments on UN pellets with 90 % density were exposed to liquid lead for one hour at T = 1 090 °C. In the case of low oxygen content in the molten lead, the UN sample remained intact, as shown in Figure 3-12. Under oxygen saturated conditions a significant surface degradation is observed, which may be attributed to formation of uranium oxides.

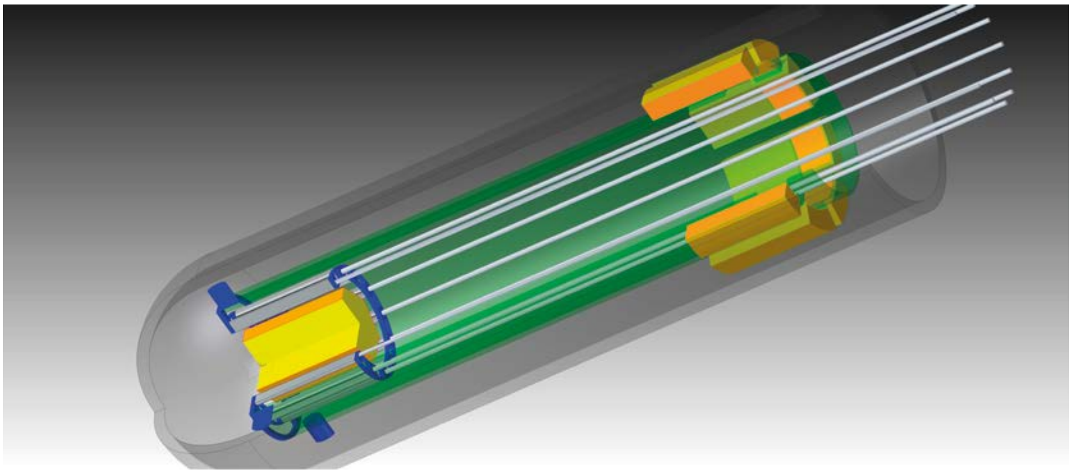

Figure 3-11. *CAD model of the primary circuit of ELECTRA.*

Figure 3-12. *UN fragment exposed to lead with low oxygen content, before (left) and after (right) the experiment.*

A thermal shock experiment was carried out, where a UN pellet of 95 % density was cooled from 1 060 °C to 200 °C by immersion into liquid lead-bismuth. The measured cooling rate of the sample surface was more than 700 K/s. As shown in image GENIUS.13, the pellet remained intact after the test, without any visible cracks.

Security

At Uppsala University, radiation detector development is carried for the purpose of allowing a combination of high resolution at high (i.e. room) temperature. The candidate materials is cadmium telluride. PhD student Anna Shepidchenko has carried out electronic structure calculations of deformed crystal lattices, using the DFT code VASP. Exercising tetragonal and hydrostatic compressions of the crystal, it was found that the tellurium anti-site defect level is shifted towards the conduction band. The magnitude of these shifts for constrictions of 1–3 % are depicted in Figure 3-14.

At Uppsala University, analysis of safeguard systems for Generation IV system facilities is made by PhD student Matilda Åberg Lindell. An assessment of the proliferation resistance of different fuel cycle options has been made using the TOPS methodology, for a small scale facility comprising of a 100 MWe LFR with on-site recycling and fuel fabrication capabilities. Comparing PUREX, PUREX+DIAMEX and a generic GANEX process, it was found that GANEX processes yield a more proliferation resistant operation, due to the absence of pure plutonium streams (Åberg Lindell et al. 2013).

Figure 3-13. *UN pellet subjected to thermal shock, before (left) and after (right) the experiment.*

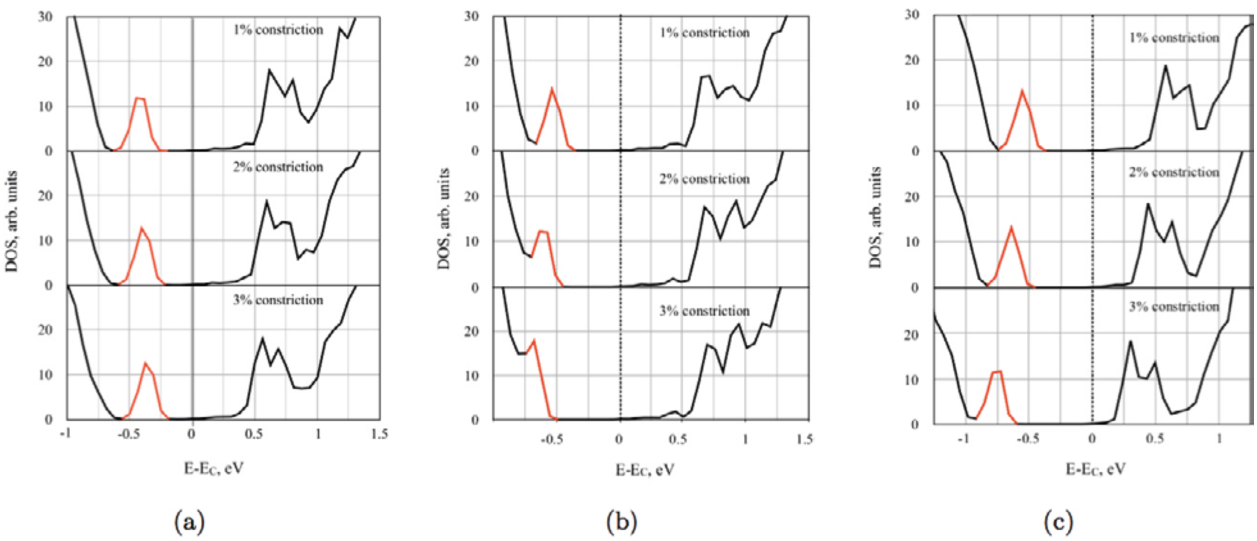

Figure 3-14. *Density of state for CdTe, for different deformations: (a): 1-D, (b): tetragonal 2-D, (c): hydrostatic 3-D. Tellurium anti-site states are denoted with red.*

3.1.2 Swedish contributions to ASTRID

Two out of the five work packages in the Swedish-French collaboration project on nuclear technology concerns research within the realm of the present report. Both are primarily focused on the ASTRID sodium-cooled fast reactor intended to be built in Marcoule near Avignon in the Rhone valley in Southern France. Two ASTRID-related projects have been awarded funding, and research work has commenced. A third project on research targeting the Jules Horowitz Reactor, being built with conventional PWR technology for materials tests is not presented here because it primarily concerns applications for present-day and near-future industrial technology. Finally, there is a work package on nuclear data research, in which an existing experimental setup for nuclear data research of ADS and fusion relevance, previously used at Uppsala University, is installed at a new neutron-beam facility under development at GANIL, Caen, France. This is presented in Section 3.5.

Safety – severe accidents

The goal of this project is to provide qualified tools for safety assessment of a sodium-cooled fast reactor, specifically the ASTRID. Development of such tools is a prerequisite for safety analysis and licensing of the new reactor. The research activities of this project will capitalize on the existing tools (computational codes and methodologies) for reactor safety analysis, including Deterministic Safety Analysis (DSA) and Probabilistic Safety Analysis (PSA) to address respectively deterministic and stochastic behaviors of nuclear power plants under abnormal (accident) conditions.

This project consists of four Work Packages, corresponding to the priority needs in the ASTRID design in terms of safety: WP1 – Core melt retention; WP2 – Simulation of severe accident; WP3 – PSA methodology; and WP4 – Analysis of severe accident scenarios.

Work Package 1: Core catcher design of SFR

The task of this work package will involve 1 (full-time) PhD student and 2 senior scientists with experience of safety analysis: 1 supervisor at KTH, and 1 supervisor at CEA, with the following activity plan:

- To acquire knowledge about the ASTRID safety design and phenomenology of severe accidents in SFRs and in LWRs.
- To identify major sources of uncertainties related to transition from LWR to SFR design and conditions to propose an approach for necessary modifications in the LWR analysis models for their adaptation to analysis of SFR phenomena.
- To provide a preliminary assessment and determine priorities for the in-depth study of one of the following severe accident phenomena in SFR: ore melt debris formation phenomena; long term coolability of the debris bed; and melt-catcher and melt-vessel wall thermo-mechanical interactions.
- Development and validation of simulation tools to fill the gaps in knowledge about SFR severe accident phenomena.
- To provide a detailed study of selected phenomena and considerations of the implications of the results for the design of ASTRID core catcher.

Work Package 2: Simulation of severe accidents

The task of this work package will involve 3 senior scientists with experience on severe accident simulation: 1 post-doctoral researcher (full-time), 1 supervisor at KTH, and 1 supervisor at CEA, with the following activity plan:

- Acquisition of the contemporary knowledge on the primary phase of a severe accident in SFRs, including experimental data, understanding and models.
- Critical review of the physical models relevant to fuel pin behaviour in the SAS-SFR code, so as to implement them in the SIMMER-III code, i.e., to extend capabilities of the SIMMER-III code to the primary phase by improving the DPIN module of the code.

- Mid-term report and peer-review publication (submitted to a journal and/or a conference) on the state-of-the-art review and modelling for the primary phase of severe accident in a sodium-cooled fast reactor.
- Qualifications of the models by performing and comparing calculations for a few suitable CABRI tests with both codes, using the existing data sets and new ones.
- New model development needs to improve the description of the fuel pin behaviour in the DPIN module of the SIMMER-III code.
- Implementation and validation of the new model(s) in the SIMMER-III code.
- Identification of future R&D needs (in terms of high-quality data for basic understanding and validation) based on the development and validation works.

Work Package 3: Probabilistic Safety Assessment (PSA) in support of the ASTRID design

The task of this work package will involve 3 senior scientists with experience of safety analysis: 1 post-doctoral researcher (full-time), 1 supervisor at KTH, and 1 supervisor at CEA, with the following activity plan:

- Acquiring general knowledge regarding PSA and operation of SFRs.
- Learning about the design features of the ASTRID reactor and the PSA computer code Risk Spectrum.
- State-of-the-art review and development of new approaches to systematic treatment of uncertainties in PSA analysis.
- Modelling of accidental scenarios (event trees) and fault trees for the systems which are involved in these scenarios. Namely the systems which are involved in Fundamental Safety Functions (reactivity control, decay heat removal) and/or support systems (electrical supply systems for instance) will be addressed.
- Simulations of accidental scenarios with deterministic thermal-hydraulic computer codes in order to evaluate the consequences of particular accidental scenarios.
- Using of the PSA model to provide insights regarding the design and safety of the reactor. Various design alternatives will be evaluated from the viewpoint of PSA in order to support the design choices of the ASTRID reactor on risk informed basis.

Work Package 4: Analysis of severe accidents scenarios

The task of this work package will involve 3 senior scientists with experience of safety analysis: 1 post-doctoral researcher (full-time), 1 supervisor at KTH, and 1 supervisor at CEA, with the following activity plan:

- Acquiring the main features of the accident behavior of a SFR (thermal-hydraulics, neutronic feedback induced reactivity effects, fuel-coolant interactions, etc.). More precise phenomena taking place in a typical severe accident of a SFR include sodium boiling, clad failure and melting, fuel melting, molten materials motion, fuel ejection, fuel dispersal and relocation and mechanical energy release. Theoretical studies and out-of-pile/in-pile experiments have been performed by CEA for conditions representative of PHENIX and SUPER-PHENIX and are available to this project.
- Learning and participation in the development of the phenomenological event trees (scenario) which is currently under development at CEA for application to ASTRID.
- Development of simplified analytical modelling for scenario calculations with deterministic multidisciplinary severe accident simulation codes.
- Use the results of the simulations to validate the analytical models on the basis of a statistical treatment of the code outputs in the full domain of variations of the input parameters of the analytical models.

- Consideration of various sources of uncertainties will be performed by propagating the sensitive parameters by means of the analytical simplified models less demanding in CPU time compared to the more detailed severe accident calculation codes (SIMMER-III for instance).
- Interpretation of the calculation results and their implication for the safety of the SFR. Specifically, the study will range from the description of the scenario and the possible branching of events through threshold phenomena up to rough assessment of radiological releases. Several stays in Grenoble will be foreseen (tutorial with a CEA expert in SFR SA modelling) in order to become familiar with the development of specific analytical tools representing the physics of the investigated scenarios.

Core physics, diagnostics and instrumentation for enhanced safety of ASTRID

This project involves researchers from Chalmers, KTH and Uppsala University. The work packages are described in more detail below.

Work Package 1. Neutronic aspects of control rod withdrawal

There are several challenges in evaluating the control rod worths of a large SFR and the aims of the project are to address the following points.

1. *Difficulties in evaluating the control rod worth and the effect of control rod withdrawal in a large SFR core.* In many cases, calculations of the worths of an individual control rod or a group of them have to be performed on a routine basis with full core transport theory models, which pose a difficulty in modelling and computation time. The traditional prediction based on diffusion theory may not give sufficient accuracy, while 3D Monte Carlo calculation is too time consuming. Moreover, estimation of a control rod withdrawal (CRW) transient and its impact on safety parameters has to be carried out in complicated 3D dynamic simulations.
2. *Uncertainties in the prediction of the worths and the life-time of the control rods.* Simple deterministic calculations predict the control rod worth with some uncertainty, and tend to overestimate the life-time of control rods. These uncertainties and biases are related to the mesh size, the energy group discretization, the homogenization of the rod etc, while the control rods also contain strong neutron absorbers with strong self-shielding effect. In ASTRID, the situation is further complicated by the pancake-like shape of the core (causing a high sensitivity to radial perturbations), the small core excess reactivity (implying limited rod insertion in the core) and by the fact that the radial power shape changes with time, as a result of the strong inner U238-to-Pu239 conversion. This makes it particularly difficult to design control rod banks that would fulfil reactivity and power shaping functions optimally at the same time. Therefore, investigation of a"robust" method for control rod worth evaluation in ASTRID-like cores is needed.
3. *Impact of control rod withdrawal on local pin power*. In large SFRs, the variation of the neutron flux is strong around the control rod positions during their insertion or withdrawal. Consequently, the power in the fuel pins around control rods may increase significantly in case of a hypothetical CRW. In safety requirement, the cladding material of the fuel pins (stainless steel) has a maximum permissible temperature of 700 °C, i.e. the maximum pin power should be kept below a safety value to avoid clad failure or fuel melting with sufficient confidence. Estimation of the impact of CRW on local pin power, and the corresponding safety margin, needs to be considered fully in a 3D dynamic problem.

Work Package 2: Acoustic leak detection

The ASTRID reactor will employ an intermediate sodium loop with several heat exchangers between sodium and the tertiary coolant, such as water and/or steam. One of the most important safety issues of SFRs is the sodium-water reaction that may take place in case of a tube failure in the heat exchanger. If not detected early enough, initially small leak may rapidly grow leading to an increase of pressure due to H_2 gas generation. This, in turn, may endanger the structural integrity of the heat removal system. Thus, as a safety precaution, several systems of early water leak detection are needed in sodium cooled reactors. The aim of the proposed project is to develop a new fast and reliable acoustic method of in-sodium water leakage detection. The method must be sensitive enough to detect leaks at early stage; however, it must be also reliable, not leading to excessive false alarms. To meet this requirement, the new method will be extensively validated against experimental data.

The proposed project will contain two parts. In the first part a theoretical assessment of the acoustic signature of the governing phenomena during in-sodium water leakage will be performed. In the second part the theoretically obtained acoustic signature will be compared against new experimental data that will be obtained at CEA. The over-all objective of the research will be to improve the sensitivity of the current acoustic detection methods, to make them more reliable and to reduce the number of spurious alarms.

The theoretical part will address the fundamental aspects of the acoustic phenomena related to the appearance and growth of a leak and to the sodium-water reactions. Dynamics of both processes will be investigated and the sources of acoustic waves will be obtained. The propagation of acoustic waves in liquid metal will be simulated using a Computational Fluid Dynamics (CFD) code. In the validation part, a model of the experimental equipment will be created within the CFD code and the calculated results will be compared with the measured data. The proposed research will help to perform an appropriate "training" of the acoustic leak detection systems prior their usage in full-scale equipment. In this way an improved sensitivity of the systems will be achieved.

Work Package 3: Advanced neutron monitoring

The key neutron sensor in an SFR is the fission chamber. Its main advantage, in comparison to other ionization chambers, is its compact size, relative insensitivity to gamma background, and its capability to monitor the neutron flux over a large dynamic range from low-flux level at startup to high-flux level at nominal power. Fission chambers can be operated in several modes: pulse mode at low count rates, Campbelling or fluctuation mode at moderate and high count rates, and current mode at high count rates only. The downside is that a specific electronic device is needed for each mode: logic circuits for pulse mode and analog-to-digital circuits for Campbelling and current mode, with different sampling rate for these two modes. This way, the measurement system is necessarily more complicated and less reliable and one has to ensure that each mode is properly calibrated with the two other ones.

It would have significant advantages if one developed a signal analysis method, based on a development of higher order Campbelling techniques, which only needed one unique type of circuit, and which would preserve the advantages of the individual detection modes. In addition of its simplicity and wide operation range, it would also make it possible to develop self-diagnosis functions. Such a so-called smart detector, capable of monitoring its own current health status, would definitely improve its maintenance and permit to schedule its replacement at reactor shutdown for refuelling. This, however, requires a substantial development of both the theory and simulation of the detection process, the associated hardware, and the signal processing methods. This is the subject of the present project.

Work Package 4: In-core detection and decision systems

Geometric perturbations and coolant void is a concern in metal-cooled reactors due to the changes in reactivity that are inferred. A classic example is the Phénix reactor, where reactivity fluctuations were encountered in 1989–1990. Coolant void, and later so-called core flowering were identified as a possible sources of these fluctuations. The purpose of the present project is to investigate how advanced in-core monitoring systems can be applied to detect and diagnose these types of irregularities in a sodium-cooled core, and the aim is to suggest a decision system, warning for irregularities.

The proposed project will investigate the added capabilities obtained when using a set of on-line monitoring fission chambers with various isotopic contents having different neutron-energy fission thresholds. Furthermore, inclusion of gamma-ray detectors will also be considered, since the response to changes in geometry and coolant void will strongly differ between gamma-rays, thermal neutrons and fast neutrons. Thus a combination of detectors may form the basis for an on-line detection and decision system in ASTRID.

3.2 KTH – reactor physics

Background

During 2010–2012 SKB has funded the division of reactor physics to conduct research on transmutation, resulting in the completion of four PhD theses. The focus of the research activities has shifted from Accelerator Driven Systems to transmutation in critical reactors, mainly Generation IV fast reactors. The potential of using existing commercial boiling water reactors for the task has also been investigated. The results of this work are presented in the present section.

In addition, research is also conducted on advanced fuel development and radiation damage physics. The outcome of this research is available in the account for the GENIUS and FAIRFUELs projects. The design and safety analysis of ELECTRA is presented in a dedicated chapter of this status-report.

Design and safety analysis of Generation IV liquid metal cooled reactors

In our previous report (Blomgren et al. 2010), it was shown that from a technological perspective, accelerator driven systems (ADS) provide the most efficient mean for transmutation of minor actinides. The ability of an ADS to accommodate large reactivity insertions makes it possible to load an ADS with ~50 % americium in the fuel, which leads to a transmutation rate of minor actinides of 42 kg/TWh. The corresponding fraction of ADS required to manage minor actinides in a fully closed fuel cycle is as small as 2–4 % (Wallenius 2011b).

Accelerator driven systems are however yet only implemented on laboratory scale, such as the GUINIVERE zero-power facility at SCK-CEN in Belgium. The capital cost of an industrial ADS producing an electric power of 130 MW has been estimated at 2.1 ± 0.2 G€ within the EUROTRANS project. This is 4–5 times higher than the specific capital cost for third generation LWRs presently under construction. The cost penalty pertaining to introducing ADSs for minor actinide transmutation may therefore be estimated at 20 ± 5 %, compared to the present open cycle in Sweden (Wallenius 2011a). This number is not insignificant. Therefore it is of interest to study in more detail how the rate of minor actinide transmutation in critical Generation IV reactors can be maximised.

At the beginning of 2010, no detailed analysis had been published on how the introduction of minor actinides affects the performance of critical Generation IV reactors under transient conditions. Vague statements claiming that 2–3 % of minor actinide loading in sodium cooled fast reactors may be permitted were used as basis of fuel cycle analyses (Tommasi et al. 1995, Wakabayashi et al. 1997, von Lensa et al. 2007).

Therefore, the reactor physics division at KTH initiated a programme with two PhD theses devoted to a consistent investigation of how much americium may in fact be incorporated in the fuel of sodium and lead cooled fast reactors featuring oxide, nitride and metallic fuels (Wallenius 2011a, Zhang et al. 2010, 2013, Zhang 2012, Tesinsky 2012, Tesinsky et al. 2012 a, b). The uranium and plutonium fractions in the fuels were adjusted to obtain a plutonium breeding ratio approximately equal to unity, in order to meet the sustainability condition of Generation IV systems.

Americium is not only highly radio-active, causing problems for fuel fabrication and spent fuel management. It also has a detrimental impact on most of the important safety parameters in nuclear reactors, such as Doppler feedback, coolant temperature coefficients and the effective delayed neutron fraction (Wallenius 2011b). Figure 3-15 shows the reduction in Doppler feedback as function of americium fraction in the fuel of a sodium cooled reactor with oxide fuel. As Doppler is the major negative temperature feedback for ceramic fuels this reduction leads to higher peak and steady state temperatures of the fuel during over power transients.

The major innovation of the KTH approach consisted of identifying how much the nominal power density (and therefore the total power of the reactor) must be reduced in order for these reactors to survive unprotected loss of flow (ULOF) and transient over power (UTOP) accidents. The modelling was made using a combination of the Serpent Monte Carlo Burnup code (Leppänen 2010) and the SAS4A/SASSYS-1 fast reactor system code (Cahalan 2001).

The major outcome of the study for sodium cooled reactors is summarised in Figure 3-16 (Zhang et al. 2010, 2013, Zhang 2012). The limiting transient is here found to be the transient over-power accident. Hence, oxide fuels are more vulnerable to the presence of americium, and the power density must be reduced by 6 % for each percent of americium that is introduced in the fuel.

Metallic fuels are less sensitive to a loss of Doppler feedback, since axial expansion of the fuel acts as a negative feedback mostly independent of americium content. Still, the relatively low melting temperature of metal alloy fuels limits the power density that can be applied. Considering nitride fuels, the combination of a high thermal conductivity with high margins to failure, make them the best choice transmutation of americium in critical Generation IV reactors. With nitride fuels, the limitation will not be the safety performance, but rather the handling of spent fuel assemblies in air (Buiron et al. 2011).

In the case of lead cooled fast reactors, metallic fuels were not investigated, as they are not compatible with lead coolants (they dissolve in lead). Again, oxide fuels feature the least favourable performance (Tesinsky et al. 2012a), whereas nitride fuel allows to load the core with a higher fraction of minor actinides (Tesinsky et al. 2012b). The power penalty pertaining to americium nitride introduction is slightly larger in lead cooled reactors (4 % per percent Am) than in sodium cooled reactors (3 % per percent Am). The reason for this is the larger coolant temperature coefficient in LFRs as compared to SFRs. This is a result of the wider fuel pin lattice spacing of lead reactors which must be applied, in order to remove heat production at a coolant velocity of 1–2 m/s, as compared to the 7–8 m/s used in SFRs.

As lead coolants behave considerably better during severe accident scenarios, the main outcome of the investigation carried out at KTH is the recommendation to use lead fast reactors with nitride fuel for transmutation of minor actinides. One may note that this is the strategic decision taken taken in Russia, where the BREST-300 reactor is slated for operation in 2020.

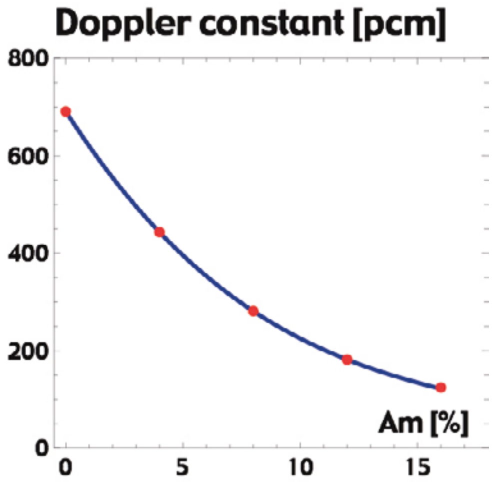

Figure 3-15. *Doppler constant as function of americium fraction in the fuel of sodium fast reactors with plutonium breeding ratio = 1.0.*

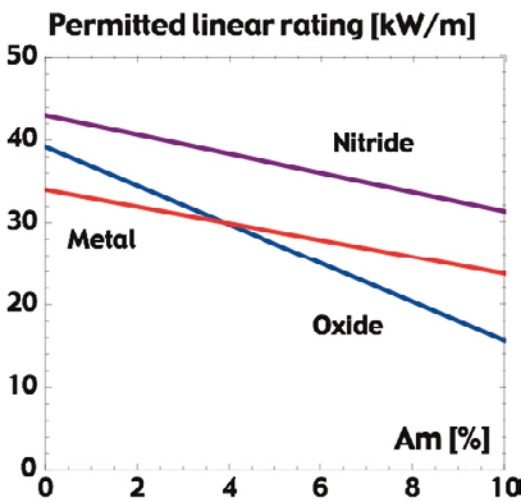

Figure 3-16. *Permitted power density in the fuel of sodium cooled fast reactors with Pu breeding ratio ≈ 1.0.*

Design and safety analysis of boiling water reactor fuels for transmutation.

Since the use of Generation IV systems for nuclear power production until recently have been expected to remain more expensive than light water reactors, KTH also has studied to what extent existing light water reactors may be used for minor actinide transmutation. Previous studies for PWRs made by CEA indicated that inventories of americium may be stabilised in a system where 40 % of all PWRs are used for recycle of this element (Youinou et al. 2003, Youinou and Vasile 2005). A major problem with this approach is the huge increase in neutron production during fuel management, due to spontaneous fission of Cf-252 produced by transmutation of curium (Salvatores et al. 2005). Therefore, recycle of curium will be extremely difficult and costly to carry out in PWRs.

BWRs feature a slightly harder neutron spectrum than PWRs, and may therefore a priori be better suited for minor actinide transmutation. Here, the coolant void worth is the limiting factor. Licensing of fuels for which it becomes positive may be difficult. Hence, a broad investigation of how much americium and curium that may be introduced into MOX fuels of BWRs while maintaining negative coolant void worth was initiated (Zakova 2012). Since the void worth will increase as plutonium quality of the MOX fuel decreases, a full study of the approach to equilibrium fuel during multi-recycling is necessary. This was carried out using the burnup-capability of MCNPX, using a full 3D-model of the Ringhals-1 BWR. The 3D-model is required in order to correctly account for effects of leakage.

The major result of the study is that the maximum permitted fraction of higher actinides is about 2.5 % in uranium based MOX fuels. This is sufficient to yield a burning rate of 3 kg/TWh of electricity production (Zakova and Wallenius 2013). An interesting outcome is that using thorium based MOX fuels, the permitted fraction increases to about 4 %, leading to a higher actinide burning rate of about 8 kg/TWh_e. Since the production rate of Am and Cm in LWRs is 3.2 kg/TWh_e, the support ratio of BWRs required to manage higher actinide production in LWRs with conventional fuels would be more than 1:1 for uranium based MOX fuels, but only 1:2 for thorium based fuels. Here it should be recalled that introduction of a thorium cycle cannot be carried out using existing industrial facilities for reprocessing of spent fuel, and would therefore become extremely costly.

Moreover, the specific neutron activity due to spontaneous fission of Cf-252 reaches 2 MBq per gram of fuel after ten years of cooling. Even though this is an order of magnitude lower than in the case of PWRs (Salvatores et al. 2005), it is two orders of magnitude above that of fast reactors carrying out recycle of curium. Thus, it is evident that fast reactors should be used for recycle of minor actinides.

3.3 Chalmers University – nuclear chemistry

During the time period 2010–2012 there have been two Ph.D. dissertations in the field of Partitioning and Transmutation (P&T) at Chalmers University of Technology (Anna Fermvik, “Radiolytic Degradation of BTBP type Molecules for Treatment of Used Nuclear Fuel by Solvent Extraction” (Fermvik 2011) and Emma Aneheim “Development of a Solvent Extraction Process for Group Actinide Recovery from Used Nuclear Fuel” (Aneheim 2012). A licentiate thesis on “Diluent and Solvent Effects in Liquid-Liquid Extraction Systems based on bis-triazine-bipyridine (BTBP)-class Ligands” was presented by Elin Löfström-Engdahl. Three diploma works have also been presented: Mikaela Holm created a computer program called RADTox for estimation of the radiotoxicity of separated and transmuted used nuclear fuel, Lovisa Bauhn made calculations for a future pilot plant for P&T recycling in Sweden, and Marcus Hedberg studied the possibilities to prevent radiolytic degradation of extractants by addition of inhibitors or scavengers.

The years 2010–2012 have also been a period of facility improvement at Chalmers. A new laboratory dedicated for the development of new Pu- and Am-based fuels has been built and made operative. It contains e.g. four new glove boxes, a line for sol-gel production of fuel material, a pellet press, a furnace for sintering and reactions up to more than 2 500 °C, a tube furnace for temperatures up to 1 500 °C, a SEM (Scanning Electron Microscope), an XRD (powder X-Ray Diffractometer), and equipment for milling and cutting fuel pellets. Two new detector systems for alpha spectroscopy have also been bought. The old ^{60}Co gamma source has been reloaded with 0.91 PBq of ^{60}Co and today it gives a dose rate of ~ 14 kGy/h.

Development of a GANEX process

A GANEX (Group ActiNide EXtraction) process consists of two liquid-liquid extraction cycles; a first cycle where the uranium bulk is removed from the fuel dissolution liquor and a second cycle (the actual GANEX extraction) where the transuranic elements as well as residual uranium are extracted together as a group, to increase proliferation resistance. In the work at Chalmers, only the second cycle has been studied. The Chalmers GANEX solvent comprises the extractants $CyMe_4$-BTBP and TBP in cyclohexanone. The GANEX research has mainly been focused towards process implementation (Aneheim 2012). This includes a continuous test in a single centrifugal contactor. These centrifugal contactor studies were carried out at Forschungszentrum Jülich, Germany. Studies have also been made regarding plutonium as well as palladium (Aneheim et al. 2012) behavior, and kinetics of actinide and lanthanide extraction (Löfström-Engdahl et al. 2011) using the GANEX solvent.

Process implementation

As a preparation for the continuous extraction experiments, a batch extraction test was performed. The extraction system was set to be the GANEX solvent as organic phase, with an aqueous phase consisting of a HAR solution (described in Aneheim 2012) with an addition of trace amounts of actinides and ^{152}Eu as well as suppressing agents for palladium, zirconium and molybdenum (bimet and mannitol), and sulfamic acid for cyclohexanone stabilization. A batch extraction process scheme can be found in Figure 3-17.

The results from the extraction and stripping using glycolic acid stripping are shown in Table 3-1 (Aneheim 2012). This experiment was successful and followed by a single centrifugal contactor step that made flow sheet calculations possible (Aneheim et al. 2012). Using the data obtained from the single centrifugal contactor could be concluded that with Θ=0.6, 15 extraction steps, 3 acid scrub steps, and 1 strip step, a high fraction of the actinides can be recovered (Aneheim 2012).

Table 3-1. An and Ln recovery in the strip solution after 7 extraction and 6 + 6 acid scrub steps and one strip step step. Θ = 0.6.

Element	% of initial
Eu	0.32
U	93.5
Np	99.8
Pu	99.6
Am	99.8

Diluent Effects

Another task has been to focus on diluent effects and therefore an old study of alcohol diluents and the BTBP ligands (Löfström-Engdahl et al. 2012b) has been followed up. A study of silver extraction in different diluents and using various BTBP ligands has also been made (Löfström-Engdahl et al. 2012b). The time needed for reaching extraction equilibrium in the Chalmers GANEX solvent has been studied in Löfström-Engdahl et al. (2011) and a literature study (Löfström-Engdahl et al. 2010) has been made.

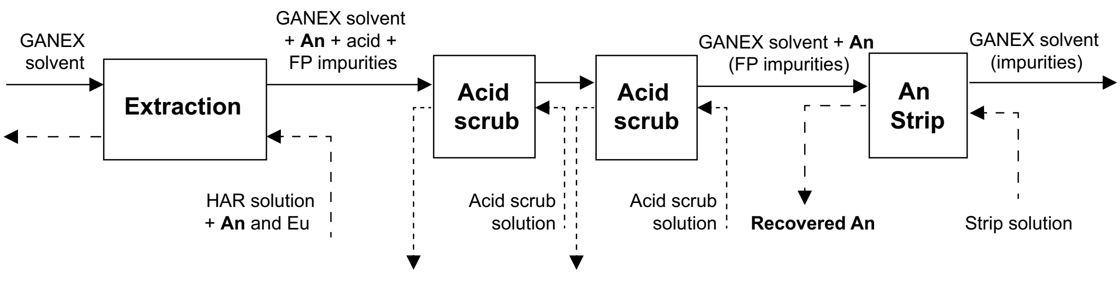

Figure 3-17. Schematic picture of the batch extraction experiments.

Re-interpretation of old data

In order to explain some of the differences in the phase contact time needed to reach extraction equilibrium in different diluents the interfacial tension of BTBP-C5 and alcohols as diluents (hexanol, heptanol, nonanol and decanol) has been determined. Interfacial tension can be compared with the more commonly used term surface tension. They both describe the force needed to create a surface of a liquid. The extraction into the four diluents has also been studied, both at equilibrium and as a function of phase contact time. The rate of extraction in the four BTBP-C5 systems has been possible to correlate with the interfacial tension in the extraction systems, see Figure 3-18. (Löfström-Engdahl et al. 2012a).

Silver Extraction

Silver co-extracts in the Chalmers GANEX solvent (Aneheim et al. 2012). It has several times been concluded that an exchange of the diluent affects the extraction of several elements when using BTBP- class ligands but whether this is the case also for silver has not been known. Therefore an exchange of the diluent cyclohexanone for octanol has been made. The slight difference between the extraction into the two diluents is in this case, however, not statistically significant (Löfström-Engdahl et al. 2012b). In the study, it was however, concluded that silver is not forming any water soluble complexes with the ligands, and that it is not extracted by the diluent itself. It was also shown that an addition of 30 % TBP to cyclohexanone does not affect the silver distribution ratio.

Solvent degradation

Separation systems for transmutation of used nuclear fuel have been shown to be sensitive to ionizing radiation. C5-BTBP and CyMe4-BTBP dissolved in cyclohexanone are less degraded by radiation than corresponding solvents with hexanol as diluent. This is explained by the generally higher radiolytic stability of cyclohexanone, and the consequently lower yield of reactive species that can react with BTBP. A more rapid degradation of C5-BTBP when an aqueous phase is present during irradiation is probably mainly caused by the many highly reactive water radiolysis products (Fermvik 2011).

The presence of nitrates in the aqueous phase results in a more oxidizing environment and consequently yields products with ketone groups instead of hydroxyl groups.

There is no clear difference between alpha and gamma radiolysis of C5-BTBP in cyclohexanone, at least at doses < 15 kGy. The same degradation products are formed, and the decrease in extraction efficiency is similar.

Some of the degradation products resulting from radiolysis of C5-BTBP extract metals.

The decreasing *SFAm/Eu* observed for C5-BTBP in several previous radiolysis studies is explained by the addition of oxygen in the degradation products, thus diminishing the difference in affinity for Am and Eu (Fermvik 2011).

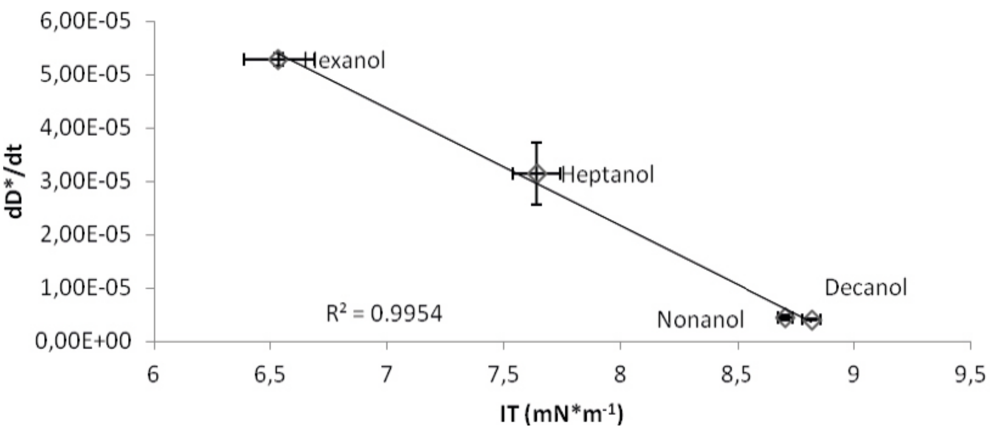

Figure 3-18. *The variation in distribution ratio as a function of interfacial tension.*

Fuel lab

Zirconium bearing microspheres as precursor material for production of zirconium nitride have been produced by the sol-gel method and characterized by SEM/EDX. The carbon additives that have been studied can be divided into two main groups. The first group contains three different kinds of elemental carbon material: carbon powder, carbon nanotubes (CNT) and carbon nanopowder (CNP). Dispersion stabilization of these three carbon species have been performed to find a sol system that could be gelled without encountering a high degree of settling of carbon material during the gelation step. The second group of carbon additives that has been tested contains material that can be chemically converted into elemental carbon. The only material that has been tested in this group is sucrose.

The zirconium containing microspheres, which can be seen in Figure 3-19, can be made without the occurrence of peptizing of the microspheres. Although when dried the microspheres displays a certain degree of cracking; this is, however, not at present considered a major problem. Both CNT and CNP seem to be usable as carbon addition. The method that has been applied to reach this stability is the addition of a nonionic surfactant as a steric stabilizer. This method seem to be applicable when working with non and/or low active material such as zirconium or natural uranium, but since the surfactant addition increases the transfer of water into the gelation oil this method may show inadequate when working with highly active substances if these to a to large degree transfer with the water into the oil. One alternative to the surfactant that will be tested in the future is the stabilization of carbon particles by addition of viscosity increasing chemicals to the feed broth such as poly vinylalcohol (PVA).

Sucrose could successfully be added as carbon containing compound during gelation but the leakage during washing seems to be quite large.

RADTox and Pilot plant calculations

To be able to estimate the radiotoxicity of the spent nuclear fuel after different treatment scenarios is of high importance. For this purpose a computer program "RADTox" has been developed. By using this program radiotoxicity curves as function of time with varying partitioning and transmutation efficiencies are produced.

The possibility of demonstrating the GANEX process on pilot plant scale using centrifuges for phase disengagement was investigated by Lovisa Bauhn. The focus of the work was process calculations, criticality calculations and estimation of radiation shielding requirements for a plant with a process capacity of approximately 1 ton of spent fuel per year.

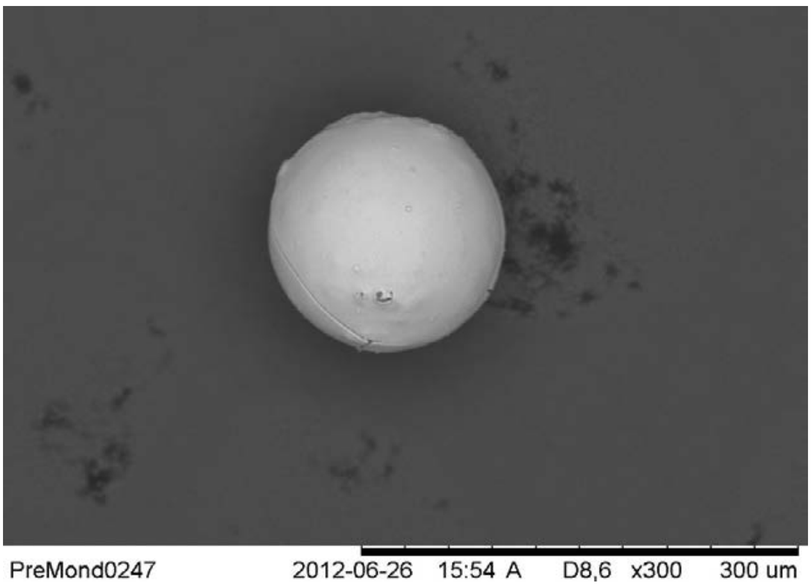

Figure 3-19. *Zirconium bearing microsphere. Picture achieved using the SEM equipment in the fuel lab.*

Calculations based on the distribution ratios for extraction by the GANEX solvent from a metal loaded aqueous phase suggest that it would be possible to achieve 99.9 % extraction of the actinides in a counter current process with 5 extraction stages.

3.4 Uppsala University – nuclear data

The applied nuclear physics division of the Department of Physics and Astronomy, Uppsala University consists of several research groups of which two perform research related to the field of separation and transmutation; the nuclear reactions research group and the nuclear fuel diagnostics and safeguards group.

The nuclear reactions group has a long standing experience in performing experimental studies at neutron beams. This group's mission is to bridge the gap between basic nuclear physics and application needs. This is done by measuring quantities like cross sections and fission yields that have been identified as important for design, optimization and safety assessment of large scale systems. The obtained data are then used guide and constrain nuclear model codes used to calculate nuclear data libraries. These libraries are at the basis of computer simulation of nuclear physics aspects of large systems like fast reactors in particular and the nuclear fuel cycle in general. Finally, the group participates in the development of novel computational methods, so-called Total Monte Carlo, for handling uncertainty propagation directly from measured nuclear data to macroscopic parameters.

The nuclear fuel diagnostics and safeguards group has a research background in spent fuel measurements, especially gamma spectroscopic and tomographic methods. Since 2009, the Division is also active within the research on future so-called Gen IV technologies.

Experimental work

The experimental activities are carried out at various facilities. The most extensively used beam was the quasi-monoenergetic neutron beam provided by The Svedberg Laboratory (TSL) of Uppsala University. There, large data sets on various reactions at high energies, 96 and 175 MeV, have been collected over the past decade. In recent years the activities have been moved towards other facilities in Europe.

3.4.1 Experimental activities

Elastic and inelastic neutron scattering

Using the SCANDAL setup (Klug et al. 2002), elastic neutron scattering off a number of targets at 96 MeV has been measured. Studied targets are H, D, C, Si, Y, Fe, Pb (Öhrn et al. 2008). The latest development concerns measurements of the (n,xn') reaction. This reaction channel is very difficult to measure and hence only one data set is available in the important intermediate energy region ranging from 30 to 200 MeV. However, good knowledge of this cross section is important for ADS designs and for benchmarking of nuclear model codes. Using the SCANDAL setup a measurement for Fe and Pb targets has been performed (Sagrado García et al. 2011). The result is somewhat surprising showing a higher than expected cross section towards lower energies for the emitted neutrons. An independent analysis that can partly verify these results is currently underway at UU (Gustavsson et al. 2014).

The SCANDAL setup was equipped with larger CsI crystals to allow unique measurements at the highest beam energy available at TSL, 175 MeV. Experimental runs using Si, Fe and Bi targets have been performed. Unfortunately, the background situation at the high energy turned out to be too serious for the SCANDAL setup and this research program has now been terminated.

Light-ion production

The production of light ions (p, d, t, ^{3}He and alpha particles) was measured with the Medley setup in a series of experiments (Pomp et al. 2010). Such data are particular important for radioprotection (tritium), and provide information on the gas production in accelerator-driven systems. After finishing the research program at 96 MeV efforts have been made to measure also at 175 MeV using the new neutron beamline developed at TSL (Prokofiev et al. 2007).

Using Medley data on light-ion production from C, O, Si, Fe, Bi and U have been collected. These nuclei are both of key importance as, e.g., construction or target materials in accelerator driven systems, and for nuclear model code development. Analysis of these data sets is or has been performed at Uppsala University, LPC Caen (France), Kyushu University (Japan), and Chiang Mai University (Thailand) (Bevilacqua 2011, Bevilacqua et al. 2011, Hirayama et al. 2011, Pomp et al. 2013).

This most recent work at TSL has been supported from Swedish industry and authority through the NEXT (2006–2010) project and the European Commission via the ANDES project. Most of the beam time at TSL for the Medley and SCANDAL runs were possible due to the financial support from the European Commission via the EFNUDAT and ERINDA projects.

Fission studies

Yields for neutron induced fission of ^{232}Th and ^{238}U at peak energies of 33, 45 and 60 MeV have been measured at the quasi-monoenergetic beam at Louvain-la-Neuve, Belgium (Simutkin 2010, Ryzhov et al. 2011). This activity has been performed in close collaboration with the Khlopin Radium Institute, St. Petersburg, Russia. A Frisch-grid Ionization Chamber and targets thin enough to allow simultaneous measurement of both fission fragments have been used. Results from the peak energies have been obtained. The measurement results allow for benchmarking of nuclear models and studying how fission yields change as a function of incoming neutron energy. Such information is highly relevant for the back-end of the fuel cycle, from both fast reactors and ADS. Currently the data from the low-energy tail are under analysis. This extends the energy range of the measurement down to 10 MeV. Results will be presented at the ND2013 conference in March 2013. This project was supported via the NEXT project.

Measurements of fission fragment properties of ^{234}U at neutron energies between 0.2 and 5 MeV have been performed at the Joint Research Centre of the European Commission in Geel, Belgium. The study of the ^{234}U(n,f) reaction is relevant even for present-day and fast reactor applications since it involves the same compound nucleus as second chance fission of ^{235}U. This work has lead to several interesting results concerning the fission process, in particular angular distributions, but also concerning experimental and analysis techniques used in this type of measurements (Al-Adili et al. 2012a, b, Al-Adili 2013, Hambsch et al. 2012).

Fission yields measurements at IGISOL

The applied nuclear physics division is involved in the so called IGISOL project, Jyväskylä. Finland, with the purpose of measuring neutron induced independent fission yields on different actinides of relevance for partition and transmutation. This research is conducted in collaboration with a group at the accelerator laboratory of the University of Jyväskylä and supported by SKB though the AlFONS project, financing two PhD students. The measurements are relevant for both the back-end of the fuel cycle and safety and economy of fast reactor operation since it aims at improving knowledge about the fuel inventory during reactor operation. They are also linked to the computational work an evaluated nuclear data described below.

The Jyväskylä group is on the forefront when it comes to accurate measurements of reaction fragments from nuclear interactions involving short lived nuclei. With the Ion Guide Isotope Separation OnLine (IGISOL) technique high yields of reaction fragments are selected and then accurately determined through mass measurement in the JYFLTRAP Penning trap. This method has proven to be very useful for the determination of independent fission yields, and experiments have been performed with proton induced fission on Th-232 and U-238 (Penttilä et al. 2010, Mattera et al. 2012).

In order to do neutron induced measurements in a relevant neutron energy spectrum, a suitable neutron source is needed. The IGISOL facility was recently supplied with a new cyclotron which will provide proton or deuteron beams of the order of 100 μA with up to 30 MeV energy. Therefore a neutron converter is being designed. Different options have been investigated, and the current approach is to utilize protons on a water cooled Beryllium plate through the Be(p,n) reaction

(Lantz et al. 2012). The design is based on simulations with Monte Carlo codes such as FLUKA and MCNPX, and deterministic codes such as COMSOL Multiphysics. There will always be uncertainties in the predictions given by the Monte Carlo codes. Therefore, a benchmark measurement was performed in June 2012 at The Svedberg Laboratory in Uppsala, funded through the ERINDA EU framework programme.

In order to obtain competitive count rates the fission targets will be placed very close to the neutron converter. The goal is to have a flexible design that will enable the use of neutron fields with different energy distributions, i.e. both thermal and fast reactor like neutron spectra. Different fields can be obtained by varying the proton energy, inserting moderator materials, using deuterons instead of protons, or by varying the converter material and thickness. Challenges for the design include the need for cooling, how to avoid too much activation of the converter material, and an intense background of low energy neutrons and high energy photons.

Activities at NFS

A high-power superconducting driver LINAG is currently under construction at GANIL, France as part of the SPIRAL-2 project. The LINAG will be used for neutron production in the Neutrons For Science, NFS, facility which, besides fundamental research, will allow research on applications like transmutation of nuclear waste and future reactor designs (Ledoux et al. 2007).

With support from the Swedish Research Council the Medley facility, previously used at TSL, will be moved to NFS for light-ion production and fission measurements (Gustavsson et al. 2012). Primarily an intense pulsed white neutron beam will be used: This beam will allow measurements with incoming neutron energies ranging approximately from 5 to 30 MeV. In a second phase also quasi-monoenergetic neutron beams in the 20–35 MeV range are expected to become available. The Medley setup will be refurbished with suitable fast detectors to allow for time-of-flight measurements with sufficient energy resolution of the incoming neutrons. For this Parallel Plate Avalanche detectors will be used. A key measurement will be simultaneous registration of elastic np scattering and fission events from a ^{238}U-CH_2-^{238}U sandwich target. This will allow for a precision measurement of the relative size of these important reference cross sections. A successful measurement will contribute to reduced cross section uncertainties of all measurements used in nuclear data evaluations done relative to the ^{238}U fission reaction.

3.4.2 Nuclear data studies

The Total Monte Carlo, TMC, technique allows for straight forward propagation of uncertainties arising from the underlying measurements to the final large scale application (Koning and Rochman 2008). This method has a potential huge impact since it offers a work-around solution for the long-standing problem of handling uncertainties and covariances by a combination of brute force computing and advanced nuclear reaction model codes. For the later the TALYS code is used (Koning et al. 2008). The nuclear reactions group has, with initial support from the Genius project funded by the Swedish Research Council, taken up research in this promising field and a close collaboration with the developers of the TMC method and the TALYS code and an adjoint professorship for Arjan Koning from NRG, The Netherlands, has been created at UU.

The TMC scheme is illustrated in Figure 3-20. Model code calculations with randomized input parameters yield results for cross sections, fission yields etc which are benchmarked versus experimental data. Accepted model calculations are passed to the next step which might, e.g., be reactor simulations or fuel inventory and burn-up calculations. This process is repeated for a large number of times and yields uncertainties on the simulation results without the need for mathematically correct uncertainty propagation and covariance calculation for the evaluated data files. The process can, however, also be used to create TALYS Evaluated Nuclear Data Libraries, TENDL (Koning et al. 2013). The later contains covariances and can be used in the more traditional way.

Current work at UU with TMC focuses on improving the TMC methodology for higher efficiency to make the process less computer intensive. This will be of high importance when full core burn-up calculations are to be made. Such an improved TMC method has been developed together with NRG (TMC-2.0), addressing the main drawback of the original TMC method, namely the time multiplication factor necessary for uncertainty propagation in the case of Monte Carlo simulations. With almost no additional calculation time, uncertainty propagation can now be achieved, thus allowing an estimation of uncertainties for any cases.

UU is also, within the Genius project, using TMC to investigate the uncertainty in the inventory due to cross-sections uncertainties in the case of ELECTRA, to investigate the uncertainty in safety parameters due to nuclear data uncertainties for ELECTRA, and improving TENDL for ^{56}Fe. An example of a macroscopic parameter distribution due to uncertainties in the underlying nuclear data is shown in Figure 3-21.

Gen IV research

Core monitoring

At present, two PhD projects are carried out on core monitoring for fast detection of possible anomalies in the core of Gen IV reactors.

Firstly, liquid-lead-cooled concepts are studied within the Swedish GENIUS program. The results indicate that the occurrence of bubbles in the core due to possible leakage in the heat exchangers, which may be foreseen to cause reactivity transients, should be detectable by monitoring changes in the neutron energy spectrum using a combination of fission chambers with ^{235}U and ^{242}Pu (Wolniewicz 2012, Wolniewicz et al. 2013).

Secondly, the concept of monitoring Gen IV cores using a combination of different detectors will be studied also for liquid-sodium-cooled cores in a newly started PhD project. These efforts are part of a large Swedish-French collaboration on research directed towards the first Gen IV prototype reactor, ASTRID, which is to be built in Cadarache, France. The PhD student, Vasudha Verma, started in November 2012 and will spend the major part of her PhD studies at CEA in France.

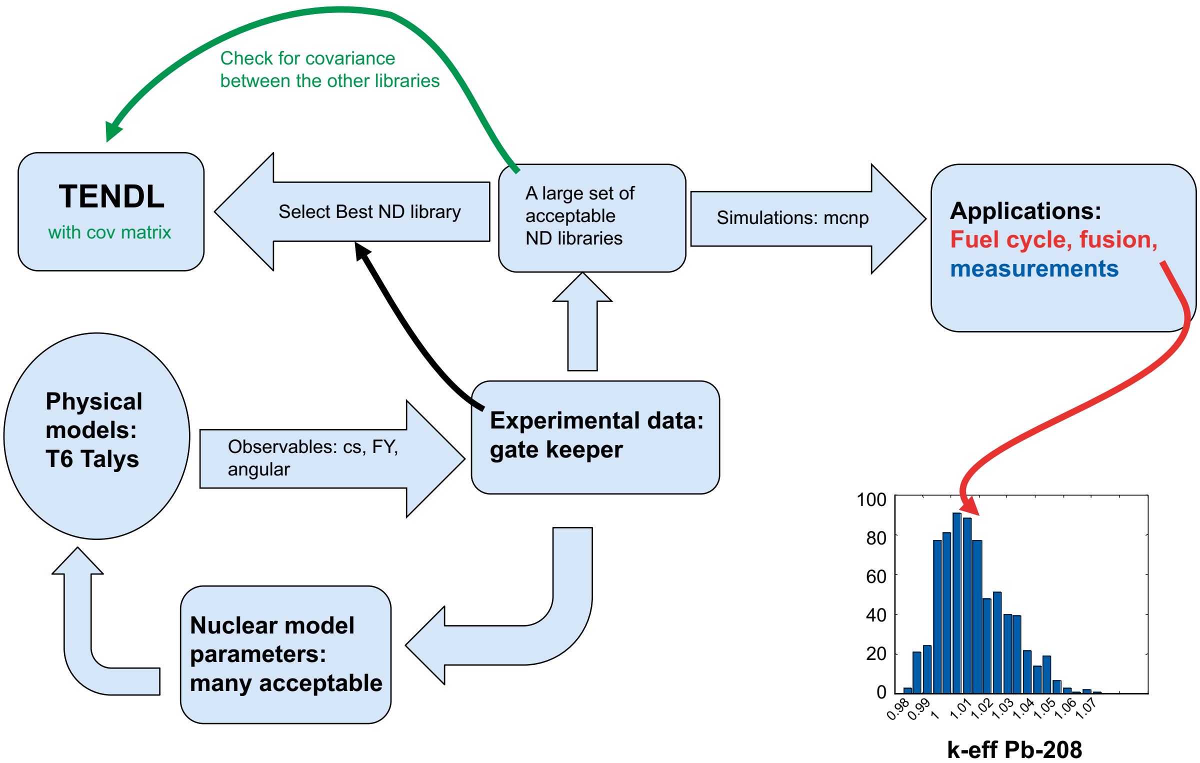

Figure 3-20. *Flow scheme of the TMC methodology. The physical models and the experimental data are at the core of the procedure. The produced evaluations are used for both the search for the best nuclear data evaluation (TENDL) and for Monte Carlo simulations of large scale systems producing probability distributions of macroscopic parameters.*

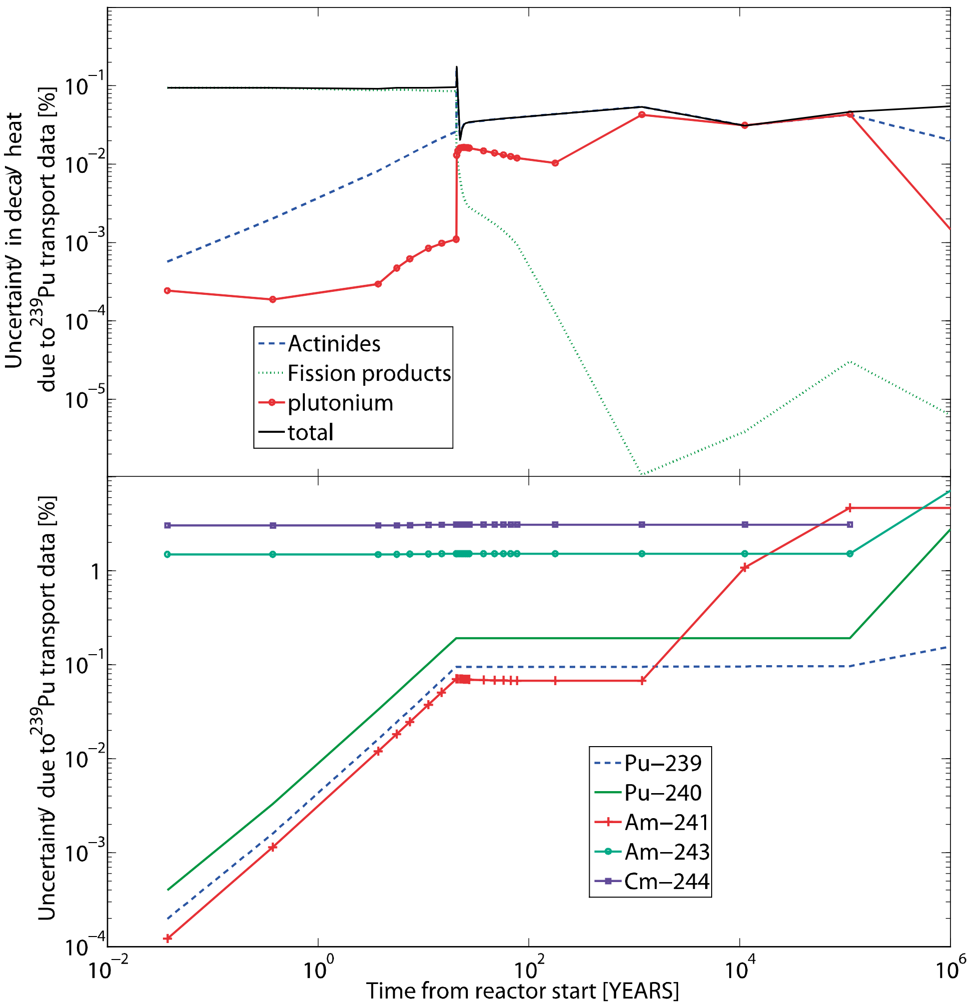

Figure 3-21. *The uncertainty in inventory and decay heat due to Pu-239 transport data as a function of timer from BOL (Sjöstand et al. 2014).*

Nuclear safeguards and non-proliferation

Also as part of the Swedish GENIUS program, the properties of lead-cooled systems is studied from the perspective of non-proliferation of potential nuclear-weapons material. The PhD student, Matilda Åberg Lindell, has submitted a paper on the proliferation assessment of the system using the so-called TOPS methodology (Åberg Lindell et al. 2013). Recently, the research has been focused on reprocessing/recycling technologies, with the purpose to identify key measurement points and suggest equipment for surveillance and inspection use that would fulfill the safeguards need of the materials and processes involved in a recycling facility.

3.5 Fusion studies for transmutation

Fusion-driven systems (FDS) for transmutation constitute an alternative to accelerator-driven systems. In mainstream ADS concepts, spallation nuclear reactions are used for neutron productions. This typically means a proton beam of the order of 1 GeV energy impinging on a heavy-element target, resulting in an energy spectrum with a peak around 1 MeV and with a tail exhibiting a 1/E intensity distribution extending up to the incident beam energy. This energy spectrum is not optimal for incineration of nuclear waste, but the interest in spallation-based neutron production is because it presently represents the beat compromise between intensity and economy.

Ideally, neutrons should have an energy exceeding the thresholds for neutron-induced fission, but not much above, because higher energies causes radiation-shielding problems. Since all actinides have fission thresholds in the range up to about 2 MeV, the ideal neutron production reaction would be the deuterium-deuterium (D-D) fusion reaction, resulting in 2.5 MeV neutrons. This reaction, however, yields a relatively modest flux, and therefore the common choice is the deuterium-tritium (D-T) fusion reaction

$$D + T \quad \rightarrow \quad He^4\ (3.5Mev) + n\,(14.1MeV)$$

that produces neutrons with the energy 14.1 MeV. This reaction has a yield two orders of magnitudes larger than the DD reaction, and the energy is still not so high that the shielding problems are paramount.

The leading development lines of controlled nuclear fusion are based on the DT reaction. The ITER tokamak device, which is under construction at Cadarache in France, will produce more than 10^{20} neutrons per second in pulses lasting for 20 minutes or more. Such a high flux is higher than required for incineration of nuclear waste. Tokamaks can only be operated in pulsed mode, but other type fusion devices (mirror machines and stellarators) can operate in continuous mode. The latter are therefore more likely to be employed in a future production stage, whereas tokamaks are the presently prime option for plasma research and technology development. The ITER device will be a large toroidal plasma confinement device, with a 36 m long plasma column with a 2 m minor radius, and it has until recently been thought of as a pure fusion device intended to produce 700 MW thermal power. Neutron sources sufficient to drive a sub-critical reactor can be substantially smaller.

The rationale for FDS is the same as for ADS. From a reactor physics perspective, the only difference Is the energy spectrum of the neutrons. Thus, the main motivation for FDS is to use it for incineration of material that is difficult to handle in large quantities in critical reactors, i.e., elements like Americium whose fission is accompanied with a very small fraction of delayed neutrons, making criticality control difficult.

The origin of the hybrid breeder reactor idea is hard to trace, since the fusion research was classified in the 1950ies. Andrei Sakharov may have been the first researcher to discuss the possibility of a hybrid breeder reactor to amplify the output of the fusion neutrons (Sakharov 1990). Hans Bethe proposed in 1978 a mirror-based hybrid reactor for production of fissile fuel (Bethe 1978, 1979), requiring a fission mantle design that slows down the neutrons to avoid plutonium burning in the mantle surrounding the fusion machine. It should be pointed out that transmutation was not the intended use of these hybrid breeder reactors. Following the proposals of Bethe, the hybrid breeder reactor studies have also aimed at breeding new fissile material U-233 from Th-232, which can be used as fuel in a separate fission reactor. After the reactor accidents in Three Mile and Chernobyl, studies on hybrid breeder reactors were essentially abandoned.

It should be noted that the global civil fusion research is dedicated to pure fusion, i.e., the aim is to reach self-burning controlled fusion plasma conditions. It would, however, be much less demanding to develop a design in which the plasma is not self-sustained, but needs to be driven, whereas still being sufficiently intense to induce fission in a surrounding blanket consisting of, e.g., natural uranium. Such a device would reach net energy output for a much smaller and less costly plasma system.

The reason for not pursuing such a development is to a large degree political. In many countries where fission has been politically incorrect, fusion has been seen as a distinctly different technology. Involving fission as the major energy production in a hybrid fusion-fission device has therefore been negatively viewed by the proponents of the idea that fusion is markedly different from conventional nuclear power.

In principle, it is possible to operate an FDS with no multiplication in the system, but this means that each fission has to be induced by an externally produced neutron, resulting in a very large neutron source and correspondingly large costs and difficulties. Most designs therefore assume that a majority of the fissions are caused by fast neutrons produced in fission chain reactions of the elements to be incinerated, like in an ADS.

In subcritical systems driven by fusion neutrons, the fission power exceeds by a large factor the power output from the fusion neutrons. As high value of the effective neutron multiplication factor k_{eff} as possible is desirable to optimize power production, whereas reactor safety is improved by lower values. Sufficient reactor safety can be expected with $k_{eff} = 0.96$, and the produced fission power then exceeds the fusion power by a factor

$$\frac{P_{fis}}{P_{fus}} \approx 150.$$

The primary fusion power is of minor importance in FDS, but the fusion neutrons control the level of fission reactions in the subcritical reactor core.

At present, the limiting factor for FDS development is the design of the plasma employed as neutron source. A number of technical solutions have been proposed, but none is close to industrial deployment. An introduction to the field and a survey of the proposed technologies has been presented in a recent SKB report (Ågren et al. 2008).

The research team points out that the present mainstream development of self-sustaining fusion by tokamaks in general and the ITER project in particular is not optimal for hybrid systems. Self-sustaining tokamaks are by necessity very large (GW scale fusion power) with neutron fluxes far exceeding the requirements for a hybrid system, with corresponding complexity and cost challenges. Moreover, tokamaks are pulsed machines, whereas a hybrid fusion-fission system better operate in steady state.

The authors present the Straight Field Line Mirror (SFLM) concept, proposed and developed by the group at Uppsala University, as a candidate for the fusion neutron source of a fuision-fission hybrid. The SFLM design consists of a comparatively small fusion neutron source (10–25 MW fusion power), aimed for 1.5 GWth power production, where the dominant part comes from fission (Ågren et al. 2010, Noack et al. 2011). The role of the fusion part is only to enhance reactor and nuclear safety and be able to burn spent nuclear fuel, including minor actinides (Noack et al. 2011). It is concluded that some extra cost will arise from the fusion neutron source, but that the price would be dramatically lower than for a fusion reactor. With such a small fusion device, the cost for the power production would come closer to that of conventional fission power.

It is anticipated that a hybrid reactor of the SFLM type would be possible to construct within a reasonably near future. Much of the required technology is already developed by the fission and fusion communities.

If a suitable fusion-based neutron source is developed, FDS might constitute an alternative to ADS. At that stage, the research challenges are essentially the same for the two concepts, i.e., materials and fuel issues can be expected to be major challenges, and the partitioning is not dependent of the core technology. The nuclear data development required is less demanding than for ADS, primarily not because the physics itself is easier, but because experiments are far less costly and the availability of experimental facilities is far better.

4 Areas for further R&D on P&T

4.1 Organizational structure of European research

The research strategy of the EU is defined in the Strategic Research Agenda (SRA) of the Sustainable Nuclear Energy Technology Platform (SNETP) (SNE-TP SRA 2013). Three major programs are outlined, NUGENIA (NUclear GENeration II & III Association) for light-water reactor technology, ENSII (European Sustainable Nuclear Industrial Initiative) for P&T and NC2I (Nuclear Cogeneration Industrial Initiative) aiming at demonstrating cogeneration of process heat and electricity based on nuclear energy.

The main objective of ESNII is to keep European leadership in fast spectrum reactor technologies aiming at high level of safety and to permit a more sustainable development of nuclear energy. With respect to the 2010 evaluation of technologies, sodium is still considered as the reference technology, since it has broader technological and reactor operations feed-back. The lead-bismuth fast reactor technology has significantly extended its technological base recently and is therefore considered as the shorter-term alternative technology, whereas the gas fast reactor technology is considered as a longer-term alternative option.

The main goal of ESNII is to design, license, construct, commission and put into operation before 2025 the sodium fast reactor prototype reactor ASTRID and the flexible fast spectrum irradiation facility MYRRHA.

ASTRID has the ambition to demonstrate capability to master the mature sodium technology with improved safety characteristics responding to societal concern of having the highest level of safety possible. Therefore, the design of ASTRID focuses on meeting the challenges in terms of industrial performance and availability, improved waste management and resource utilisation and a safety level compatible with WENRA objectives for new nuclear builds and aiming at achieving the Generation IV goals. An associated R&D programme will continue to accompany and support the development of ASTRID to increase the lines of defence and robustness of the safety demonstration of this technology, and allow the goals of the Generation IV to be reached, not only on safety and proliferation resistance, but also on economy and sustainability.

With sucessful deployment of MYRRHA, Europe could again possess a flexible fast spectrum irradiation facility that is a prerequisite for further innovations in fuels, materials and components for fast spectrum reactors. That would further diversify Europe's expertise in fast reactor technology and allow major innovations towards even more economic, sustainable and safe reactor concepts. Since MYRRHA will be conceived as an Accelerator Driven System, it will be able to demonstrate this technology, thereby allowing the technical feasibility of one of the key components in the double strata strategy for high-level waste transmutation to be evaluated. Due to the fact that MYRRHA will be based on heavy liquid metal technology (namely lead-bismuth eutectic), it can serve the role of Lead Fast Reactor European Technology Pilot Plant (ETPP) as identified in the LFR roadmap. An associated R&D programme will accompany and support the development of MYRRHA.

For the financing of the total investment cost of these facilities, it will be of paramount importance to establish the appropriate consortium structure and legal basis, allowing candidate consortium members to identify the added value of the facility for their own interest.

In parallel to the realisation of ASTRID and MYRRHA, activities directed towards the Lead Fast Reactor technology and the Gas Fast Reactor technology are envisaged. For the development of the Lead-cooled Fast Reactor, maximum synergy of activities will be sought with the MYRRHA development to optimise resources and planning. For the LFR demonstrator ALFRED, the main focus should be on design activities typical for a critical power reactor connected to the grid, as well as on R&D activities on the lead coolant, addressing the specific characteristics that differ from lead bismuth. Design activities and support R&D should be performed in the coming years to the maximum extent compatible with available resources and taking full advantage from feedback, where applicable, from the ongoing design of MYRRHA and related R&D programmes. These activities will allow the LFR consortium to reach the level of maturity needed to start the licensing phase and then the construction of ALFRED, provided that adequate financial resources are made available.

In addition to the closure of the nuclear fuel cycle in a sustainable manner, the Gas Fast Reactor has the potential to deliver high temperature heat at 800 °C for chemical process, production of hydrogen, synthetic fuels, etc. The Helium-cooled Fast Reactor is an innovative nuclear system having attractive features due to the fact that helium is transparent to neutrons and is not chemically reactive. Its viability is, however, essentially based on two main challenges. First, the development and qualification of an innovative fuel type that can withstand the irradiation, temperature and pressure conditions put forward for the GFR concept. Secondly, a high intrinsic safety level will need to be demonstrated for this GFR concept. This will imply dedicated design activities followed probably by out-of-pile demonstration experiments. These high priority R&D activities should be embedded into an overall R&D roadmap in support of the development of the Gas Fast Reactor concept. For the development, guidance and implementation of this R&D effort, a GFR centre of excellence is envisaged. It might open up the technical capability to launch the ALLEGRO gas cooled demonstrator.

Based on the ADRIANA project, a number of supporting facilities for the different systems and technologies have been identified, in addition to experimental reactors like JHR and PALLAS. The realisation and operation of these supporting facilities, in particular a fast reactor MOX production line, will be of primary importance to reach the aforementioned objectives. Maximum synergies between the different Liquid Metal Reactor technologies should be exploited, for instance in the field of instrumentation and thermal-hydraulics. Raising the financial resources to deliver the ESNII projects and build the different facilities will be a key factor of success. In this respect, international collaboration through GIF and bilateral or multi-lateral frameworks will be looked for to optimise resources.

4.2 Transmutation in critical reactors

As described above, the research on transmutation in critical reactors in Europe essentially focuses on three fast-reactor concepts; sodium-cooled (ASTRID), lead-cooled (ALFRED) and gas-cooled (ALLEGRO). The latter has, however, limited attention on P&T, but is rather targeting heat production. The texts in this section are to a large degree imported from the recent update of the SNETP Strategic Research Agenda (SNE-TP SRA 2013).

4.2.1 ASTRID

ASTRID will be a pool type, sodium cooled fast reactor of 1 500 MWth, generating about 600 MWe. ASTRID will also have provisions for experiments on transmutation of minor actinides (mainly americium) in significant quantities to allow optimised management of wastes.

ASTRID technology benefits also from the large fuel qualification database obtained from former reactors. The ASTRID start-up core will be based on MOX fuel with 15/15Ti AIM1 alloy cladding that was irradiated in the Phénix reactor and showed good performance. After Phénix's definitive shutdown in 2009, a dedicated programme of post-irradiation examination has been developed for the coming years so that the most up-to-date knowledge on AIM1 cladding will be available. This examination programme will also give feedback on the heterogeneous core concept. The material for the hexagonal tubes will be EM10, used in the last Phénix core batches and well qualified. To ensure waste reduction capability, ASTRID will continue the demonstration at higher scales of minor actinide transmutation (mainly americium) that was started at experimental scale with Phénix.

A number of important challenges have been identified:

Core meltdown probability at the lowest achievable

Initially conceived with the intention of significantly reducing the sodium void effect in case of sodium boiling concept, the CFV core concept focuses on optimising the core neutron feedback parameters (reactivity coefficients) so as to obtain improved natural core behaviour during accidental conditions leading to overall core heating. More specifically, the reactivity effect associated with sodium expansion achieved by design (sodium plenum and heterogeneous fertile plate) is negative in the event of a total loss of primary coolant, and can result in an overall negative void effect if a boiling phase is reached.

The CFV concept also shows a low reactivity loss per cycle due to the large diameter fuel pins. This geometry leads also to longer cycles and fuel residence times, as well as improved behaviour during an accidental control rod ejection transient (no ejection in SFR but inadvertent control rod withdrawal) with respect to conventional core designs. These characteristics of the CFV core concept remain to be confirmed by future simulation and experimental validation.

ASTRID will be equipped with additional safety devices to enhance the robustness of some safety margins. One example is a passive-type emergency shutdown system called SEPIA. Further R&D will be devoted to analyse alternative systems, including some ideas proposed during the EFR project.

Significant R&D efforts are ongoing and will be increased to improve instrumentation and measurement systems to reinforce core and reactor monitoring. Prevention will be consolidated by making sure all components important for safety can be inspected, as well as components capable of impacting these safety-important components. This first and foremost concerns the internal structures of the reactor block, particularly the core support and core cover plug for which efficient inspection methods must be qualified. The choice between the different reactor block internal structures described further takes into account this inspection criterion.

Decay heat removal (DHR) systems will be redundant and diversified with the ambition that the practical elimination of their total failure during a given duration can be demonstrated.

Development of dedicated DHR systems using structures for degraded situations is to be strengthened in order to have a diversified DHR system, as usual systems pass through the roof slab; such a design option needs to be studied in relation to the concept of a core catcher external to the primary vessel.

Resistance to a potential mechanical energy release accident

For the safety demonstration, in particular prevention, a core catcher will be installed in ASTRID and will be designed to recover the entire core, maintain the corium in a sub-critical state while ensuring its long-term cooling, as well as being inspectable. Several options need to be investigated as to possible core-catcher technologies, locations (in-vessel or outside the vessel) and performances.

In compliance with the WENRA approach on the independence of lines of defence, the containment will be designed to resist mechanical energy release with the objective that no countermeasures are necessary outside the site boundary in the event of an accident. R&D will be needed to support the demonstration.

Inspecting structures in sodium

Contrary to the previous Phénix and Superphénix reactors, periodic inspection of the reactor block internal structures has been integrated into the design. Although some technologies now exist that enable this inspection either from outside or inside the vessel, further R&D on optical and ultrasonic systems will be necessary to develop and select the most suitable technology to be used in the primary system.

Reduction of risks associated with the affinity between sodium and oxygen

To improve the safety and acceptability of the reactor with the de facto elimination of risks associated with sodium-water reactions, an innovative energy conversion system is being considered that uses gas for the thermodynamic transformations (Brayton cycle). This type of system has been studied by CEA in the past and has been adapted to the pressure and power ranges required in ASTRID. Further work is needed to couple this concept to the reactor through an intermediate sodium system, in order to exclude risks of gas entrainment into the core.

In case the water-steam cycle is to be retained, further improvements through R&D are to be considered on:

- Modular steam generators, whose size guarantees the integrity of the intermediate heat exchanger, and thus protects the primary system, the secondary system and the steam generator casing, even in the event of the sudden and simultaneous failure of all the steam generator module tubes.
- Steam generator concepts that ensure better protection against propagation of tube failure in case of sodium-water reaction.

4.2.2 ALFRED

The conceptual design of ALFRED (Advanced Lead Fast Reactor European Demonstrator) has been carried out as part of the 7th FP LEADER project. The work capitalises on achievements of previous FWP projects on heavy liquid metal cooled fast reactor technologies, such as ELSY, GETMAT, and EUROTRANS. Moreover, synergies between the ITER programme and the LFR R&D needs on coolant chemistry and material compatibility are under consideration. In addition, within the frame of the Generation IV International Forum (GIF), international contacts have been established with the developers of the Russian BREST-300 demonstrator and of the US SSTAR concept in order to investigate further synergies and wider cooperation.

Objectives

The Lead Fast Reactor technology is a very promising candidate among the Generation IV Fast Reactors concepts, strictly fulfilling all the main goals as defined by the Generation IV International Forum (GIF). LFR is based on a closed fuel cycle (Sustainability), the inert nature of the coolant provides important design simplification (Economics) and allows for designing decay heat removal systems based on well-known light water technology and passive features (Safety). Moreover, the LFR fuel taken as a reference in the European development programme constitutes a very unattractive route for diversion or theft of weapons-grade materials and provides increased physical protection against acts of terrorism (Nonproliferation and Physical Protection). In the last decades, the unavailability of qualified materials in a heavy liquid metal environment at relatively high temperatures (above 500–550 °C) has forced the selection of Lead-Bismuth Eutectic (LBE) as primary coolant for the ADS technology, leading to the MYRRHA project, as the European Technology Pilot Plant (ETPP) among the ESNII initiatives.

However, the objectives of large scale, sustainable and competitive nuclear energy production are only achievable through a pure lead cooled fast reactor with higher operational temperatures and higher efficiency. Long-term European LFR development will benefit from the safety features already developed for both the MYRRHA and ALFRED projects where the inertness and intrinsic characteristics of the heavy liquid metal coolants have been and will be duly taken into account through specific design provisions. ALFRED represents the first step of the LFR initiative, the European Technology Demonstrator Reactor (ETDR) of the LFR technology, the first plant connected to the grid and fulfilling the Generation IV goals. This would be the first time that a critical heavy liquid metal cooled reactor would provide electricity to the grid.

Besides the different objectives of MYRRHA and ALFRED, it is important to stress the obvious strong synergies characterised by the basic similarities of coolant technologies and further enhanced by the strong collaboration already existing among research centres and industries widely involved in both the ADS and LFR activities as well as in fusion technology. Indeed, the LFR roadmap is based on a number of European experimental facilities dedicated to Lead and Lead-Bismuth technology, and takes advantage of the nuclear data collection and operational experience gained at the Guinevere facility. As fully described below, the high level of flexibility reached in the design phase of ALFRED, will allow for a short-term deployment strategy as soon as financial instruments are available. In the LFR long term deployment strategy, the ALFRED operation will take full advantage of the gained experience and of the data made available by the above mentioned facilities, including MYRRHA.

State of the Art

Starting from April 2010 the LEADER project carried out an important set of activities with two main goals: the advancement of the conceptual design of the industrial size plant to the present European LFR configuration (ELFR), rated at 600 MWe, and the development of the design of the LFR demonstrator ALFRED, a fundamental step on the LFR roadmap. Main features of the ALFRED design are:

- pool type configuration characterised by a reactor vessel and the cavity liner safety vessel,
- hexagonal wrapped fuel assemblies extended to cover gas to simplify fuel handling (FAs weighted down by tungsten ballast for refueling and kept in position by upper grid springs during operation),

- mechanical pumps,
- double-walled straight SG tubes with continuous monitoring of tube leakages,
- reference thermal power of 300 MWth.

The thermal cycle is completely consistent with the ELFR thermal cycle: primary lead temperature being between 400–480 °C, secondary side pressure 180 bar, once-through SGs with water/steam temperature range from 335 to 450 °C in superheated conditions, the overall efficiency has been evaluated higher than 42 %.

ALFRED will also allow for testing the connection to the electrical grid, with a generated power of about 120 MWe. The safety of ALFRED is extensively based on the use of the defence in depth criteria, enhanced by the use of passive safety systems (actively actuated through locally stored energy source, always available, and fully passively operated). Safety features of the LFR system have been designed since the beginning of the activities to face challenging conditions and events, thanks to the very forgiving and benign characteristics of the coolant. As an example, there is no need for off-site or emergency AC electrical power supply to manage the design basis accident conditions, the only action needed is the addition of water to maintain the level in the decay heat removal (DHR) pools which are already sized to guarantee at least three days of unassisted fully passive operation and can be easily refilled in the following days.

The operation will take place in two main steps: the first one will be carried out with the available fuel at the time of plant start-up (present choice is MOX), the second one will exploit results of investigations carried out by MYRRHA to implement innovative fuel for the LFR demonstrator reactor. The almost parallel development of the LBE and pure-lead technologies, combined with the two-step approach foreseen for the operation of ALFRED, will allow for a more efficient exploitation of the synergies between ADSs and LFRs, as well as for a broad cooperation and related technologies spin-off. Such a target is considered technically feasible but obviously needs the allocation of appropriate financial and technical (man-power) resources. First efforts have been carried out in the past years to provide the necessary basic steps for ALFRED development, namely:

- The activities carried out by the LEADER project related to ALFRED conceptual design.
- The 2011 Romanian proposal to include ALFRED in the country's energy strategy.
- The signature of a Memorandum of Understanding in 2012 by major Italian industry (ANSALDO) and research organisation (ENEA) and the Romanian Research Institute (INR) dedicated to the development of an organisational framework for the ALFRED consortium.

The next step is the formation of an International consortium to advance both ALFRED design and licensing activities. Besides the coordination activities for ALFRED some technological development to reach a higher level of maturity of the LFR development is still needed. The related activities are summarised below, on the basis of the categorisation developed by the Generation IV LFR System Steering Committee:

- system design and component development,
- qualification of materials and development of,
- lead technology,
- innovative fuels and fuel cycle.

Challenges

- System design and component development.
- The main goals of the LFR system design and component development are:
  - approach to control corrosion and erosion of structural materials,
  - seismic isolators to cope with the large mass of lead,
  - in-service inspection techniques,
  - refuelling operations at high temperature (400 °C),
  - management of Steam Generator Tube Rupture inside the primary system,
  - prevention of freezing of coolant during all operational states.

Research on phenomena of corrosion and erosion by molten lead and their prevention for candidate structural steels for the primary system is essential. For near term deployment, the use of existing, qualified industrial materials for most parts of the reactor equipment is possible, by limiting the core outlet temperature, whereas new materials or specific coatings are being developed for special components, especially claddings. For longer term deployment, approaches beyond the usual "oxygen control strategy" may be explored to extend the operability conditions in terms of temperature range.

The fuel assemblies are fitted with an extended stem which allows the fuel handling machine to operate in the cover gas under full visibility conditions. This completely eliminates in-vessel fuel transfer equipment. Steam Generator Tube Rupture is being investigated by experimental tests aimed to demonstrate that such events do not compromise safety, i.e. they can be adequately prevented or mitigated, and will not represent a challenge to the investment protection.

Materials qualification and lead technology development

The strategy consists of two tracks, for short and medium-term deployment:

- use of existing qualified materials (short term),
- development of the coolant oxygen control for very large pools (short term),
- development of innovative materials and coolant technology (medium term),
- assessment of the possibility for application of material surface coating (medium term).

Due to the large database available, austenitic steels, and especially those of low-carbon grade, are candidates for components operating at relatively low temperatures and low irradiation fluence, e.g. the reactor vessel. Advanced austenitic stainless steel (such as the 15-15 Ti strengthened and its evolutions) in the short and medium term appear to be the most suitable solution for fuel cladding because already proven in SFRs, even if the corrosion resistance in lead still needs to be addressed. The possibility to adopt a coating for this aim is under investigation even if its performance has to be proven by irradiation tests in a lead environment.

For long term deployment, ferritic-martensitic steels appear to be among the best candidate materials for fuel cladding and structures because of their higher resistance against swelling under high fast neutron fluence. Nevertheless several characteristics of ferritic-martensitic steels such as the fatigue softening, DBTT shift under irradiation, type IV weld cracking, creep resistance and thermal ageing still have to be properly addressed.

The resulting R&D needs consist of the qualification of:

- an austenitic steel for the reactor vessel,
- lead corrosion resistant material for the steam generators tubing,
- protective coating for fuel cladding and fuel element structural parts,
- special materials or coatings for the impeller of the mechanical pumps.

The use of lead coolant implies also:

- development and validation of a technique for lead purification (prior to use and online during operation to recover activated corrosion products or e.g. volatile Hg),
- development and calibration of instrumentation operating in lead and under irradiation (fuel cladding detection instrumentation, coolant chemistry control, thermal-hydraulics monitoring, ultrasonic instrumentation under liquid metal…),
- development of techniques and instrumentation for in-service inspection,
- development of a waste management strategy for used lead.

Thus, the strategy for material qualification and lead technology development consists of a two-step approach based on the need to achieve demonstration in the short term and optimise the system for long-term industrial deployment. Consequently, as mentioned for MYRRHA, ALFRED will use already available technology (e.g. 15-15Ti for the cladding without or with surface coating) while the ELFR can fully exploit the advantages of innovative materials (innovative austenitic steels, T91, or ODS – which however still needs irradiation qualification) and of possible innovative breakthroughs regarding lead coolant chemistry and its purification. Special attention is presently dedicated, for this purpose, to the austenitic and ferritic-martensitic steels containing Al and/or Si as alloying elements, due to their high resistance to lead corrosion, even though special attention will be given to irradiation embrittlement.

Innovative fuels and fuel cycle

Fuel development from the demonstration phase to industrial deployment:

- ready-to-use technical solutions for demonstration in the near term,
- mid-term goal to confirm the use of MA bearing fuels,
- development of innovative fuel solutions for industrial deployment.

In the near term, an essential step of the LFR development is the availability of ready-to-use technical solutions, so that fuel can be provided on time tested, and qualified, with the parallel development of suitable performance codes.

In the mid-term, it is necessary to confirm the possibility of using advanced Minor Actinide bearing fuels and the possibility of achieving high fuel burn-up. In the long term, it is important to confirm the potential for industrial deployment of advanced MA-bearing fuels and the possibility of using innovative fuels having higher conductivity and lower swelling that can withstand high temperatures thus increasing fuel safety margin.

The R&D programme presents strong synergies with the SFR fuel development activities, as recognised by both communities. Due to the large number of similarities and common activities, the identification of a common line of development for both systems is of mutual interest and, consequently, strongly suggested. Qualification of the cladding material, being a very long and expensive task, may also take advantage of the research programmes carried out for both SFR and LBE systems resulting in very important savings in terms of overall cost and efficiency. Other cross-cutting research activities have been identified in the fields of core safety, fuel safety, seismic studies as well as instrumentation, inspection and repair techniques.

4.2.3 ALLEGRO

ALLEGRO is the Gas cooled Fast Reactor (GFR) demonstrator as identified in the SNETP roadmap for the development of the Gas Fast Reactor technology. (SNE-TP SRA 2013).

Objectives

The GFR cooled by helium is proposed as a longer term alternative to sodium cooled fast reactors (SFR). As well as offering the advantages of improved inspection, simplified coolant handling and low void reactivity, the GFR offers the unique advantage of fulfilling two missions:

1. closure of the nuclear fuel cycle and simultaneously providing a sustainable nuclear energy source as with other ESNII concepts,
2. delivery of high temperature heat at ~800 °C (process heat, production of hydrogen, synthetic fuels…).

The helium cooled Fast Reactor is an innovative nuclear system having attractive features: helium is transparent to neutrons and is not chemically reactive. Its viability is essentially based on:

- the development of a refractory and dense fuel,
- robust management of accidental transients, especially after the Fukushima accident.

For GFR to become an industrial reality, an intermediate objective is the design and construction of a small demonstration reactor. This reactor has been named ALLEGRO and its role, apart from being the world's first gas cooled fast reactor, consists of the following:

- pilot scale demonstration of GFR-specific safety systems taking benefit from simpler in-service inspection and repair and coolant management,

- final qualification of the innovative high temperature (ceramic) fuel at the full core level required for GFR,
- testing of the GFR-related technologies such as e.g. refuelling, spent fuel reprocessing and refabricating, helium purification & regeneration, high-temperature materials, GFR-related components,
- potential test capacity of high temperature components or heat processes.

Present status

A carefully planned and extensive R&D of GFRs started after 2001 in France at CEA and continued on the GFR 2400 MWt and ALLEGRO 75 MWt concepts until 2009, when the GFR programme was reduced. International collaboration of CEA with other European institutions took place (or is still underway) mainly within the EURATOM Framework Programmes (FP6 GCFR STREP, FP7 GoFastR). In 2010, three research institutes from the Czech Republic, Hungary and Slovakia, stepped into the ALLEGRO development, with the aim of creating an ALLEGRO Consortium and hosting the demonstrator in one of these countries. A Memorandum of Understanding was signed on 20 May, 2010 between ÚJV Řež, a.s. (Czech Republic), MTA-EK Budapest (Hungary) and VUJE, a.s. (Slovakia). The National Centre of Nuclear Research (NCBJ) Warsaw (Poland) signed the Memorandum of Understanding in 2012 as associated member.

The CEA contributes to the preliminary phase of the project. Consecutively the formation of the international Consortium is underway. The Consortium members agree to use their own financial resources in combination with the expected governmental support in their countries and international support from the EU Framework. The Consortium assumes the establishment of a GFR Research Centre of Excellence. It is worth mentioning that the Czech and Slovak republic (former Czechoslovakia) built and operated a gas (CO_2) cooled heavy-water moderated nuclear reactor KS-150 in the period 1972–1977. The demonstration of the GFR technology assumes that the basic features of the 2400 MWt GFR reactor can be tested in the 75 MWt ALLEGRO. Therefore, most of the main parameters of both reactors are similar to each other (power density, etc.).

The current CEA Concept of ALLEGRO is characterised by a metallic reactor pressure vessel (RPV), upward core cooling, control rod mechanism through the RPV bottom entry, two primary loops (realised as a coaxial cross-duct) and two circuits (the primary helium and the secondary water). Three decay heat removal (DHR) loops containing water-cooled heat exchangers located well above the core represent another important feature of the Concept.

As the production of electricity is not the primary goal, the CEA Concept has no power conversion system. The primary circuit is filled with helium pressurised to 7 MPa. The whole primary circuit is integrated, including the DHR loops, in a cylindrical close containment call the guard vessel, which is filled with atmospheric nitrogen.

The water in the secondary circuit is pressurised to 6.5 MPa. A gas/gas heat exchanger is planned for extraction of process heat. Since ALLEGRO will be a demonstrator of the GFR2400 concept, the development, testing, and qualification of the advanced fuel (applicable in the GFR2400) is of primary importance.

Two successive core configurations are, therefore, expected. The starting core will be based on MOX fuel containing ~25 % Pu in stainless steel cladding. This fuel will be derived from the SFR programme and will serve as a driving core (Tinlet/Toutlet He 260/530 °C) for six experimental fuel assemblies containing the advanced ceramic fuel (pin-type mixed carbide fuel in SiCf/SiC claddings resistant in accident conditions up to 1600 °C for few hours). Flow reduction in these assemblies will enable to reach ~850 °C at the outlet from these assemblies. The pressure drop in the core is designed to be below 0.15 MPa to ease the gas circulation. The final core of ALLEGRO will consist solely of the ceramic fuel and will enable the operation of ALLEGRO at the high target temperature (Tinlet/Toutlet He 400/850 °C).

To maximise the similarity with the GFR2400 and to increase safety, the Consortium proposed an intermediate gas circuit filled with He+N2 mixture to insert into the scheme of the CEA Concept. The second reason was to minimise the risk of a massive water leakage into the primary circuit filled with hot helium (corrosion, criticality). A gas expansion turbine was proposed for this circuit by CEA as a safety feature just to produce power for the blowers in case of a blackout. The third circuit, conducting the heat to the cooling tower contains alternatively a gas/water heat exchanger or a steam generator for a power conversion system represented by a small steam turbine, operated in a Rankine cycle. The use of both the steam turbine as well as the gas expansion turbine for ALLEGRO is still under discussion. To support the theoretical R&D, several experimental facilities were designed at CEA. Some of them were constructed and started to generate data already before 2009. An integral high-temperature helium loop for testing of the DHR system, components and code validation, however, has not yet been built. The existing experimental facilities for GFR-related research were summarised within the FP7 project ADRIANA in 2011.

Challenges

The challenges are mainly related to the demonstration of safety, the fuel technology able to withstand high temperatures, the material issues, and the helium-related technology.

Optimisation of the design for ensuring cooling of the core in accident conditions

During a loss of external power for blowers in the gas circuits (especially during loss-of-primarycoolant accidents associated with significant depressurisation), forced convection is required for successful removal of the decay heat. CEA proposed to take advantage of the above mentioned gas turbine in the intermediate He+N2 circuit driven by the decay heat transferred from the primary circuit through the gas/gas heat exchanger into the intermediate circuit. This turbine is expected to drive the blowers in both gas circuits in accident conditions. Analyses of this option as well as the assessment of technological feasibility have to be performed.

Qualification of the GFR-related DHR approach

The behaviour of the GFR-specific DHR system, i.e. additional water-cooled heat exchangers located in loops well above the core, was simulated numerically but not yet experimentally. An integral loop for a complex test of the DHR approach is planned at CV Řež (Czech Republic) to test the following phenomena:

- qualification of the GFR-related DHR approach,
- validation of system codes (e.g. CATHARE or those under development in HTR projects),
- capability to switch the core cooling from the main loops to the DHR loops and their capability to operate in natural circulation in expected conditions,
- capability to avoid core by-pass in LOCA conditions especially under interaction of several main and DHR loops.

Development of the carbide (U,Pu)C fuel in SiCf/SiC cladding for the second core

CEA evaluated this type of fuel as a promising option for the high-temperature GFR core and achieved a significant progress in its development. Priority was given to pin-type fuel. The plate-fuel, originally considered as a very promising option, was abandoned. The following R&D is expected:

- further optimisation of the SiCf/SiC design, properties, performance, cost, and technology (component production route, plug technology, hermetic sealing using a suitable liner material, irradiation-enhanced creep, oxidation by impurities in the helium coolant, embrittlement by irradiation at low temperatures),
- out-of pile and in-pile testing of the mixed carbide fuel & SiCf/SiC segments (optimisation and assessment of porosity, thermal conductivity, etc.),
- minimisation of the fuel-cladding mechanical & chemical interaction,
- assessment of the SiCf/SiC abrasion/erosion in flowing helium containing impurities,
- development and validation of the models for swelling and fission gas release from the mixed carbide fuel in operational conditions,
- implementation of mechanical & physical properties into fuel behaviour codes and their validation for the planned operating domain with respect to temperatures, burn-up, etc.

Mitigation of severe accidents (SA)

Efficient design features for mitigation of SA are expected to be proposed and implemented into the ALLEGRO concept. Thorough analyses using suitable severe accident codes should prove that WENRA requirements would be fulfilled for both the MOX and the ceramic cores. This concerns at present e.g. the assessment of so called unprotected accidents. The behaviour of the molten fuel and its coolability, have to be studied.

Development & qualification of the wire-wrapped MOX fuel for the first core

The MOX fuel from the SFR programme needs to be fabricated and qualified for its use in helium coolant with prototypical GFR parameters. This includes also the optimisation of both the cladding material and the hexagonal wrapper tube as well as the extension of the fuel performance codes to the ALLEGRO fuel (validation of the fission gas release model, assessment of temperature non-uniformities in the MOX bundle, heat exchanges, pressure drop, provision of various physical properties e.g. heat transfer into helium, material properties of both MOX and cladding).

Development of thermal barriers and insulation materials

Thermal barriers are needed in the GFR design to protect structural metallic materials from excessive temperature load:

- thermal barrier (panels) protecting the inner surface of the reactor pressure vessel (goal to withstand short term 1 250 °C in accident conditions),
- thermal shield of the experimental (U,Pu)C assemblies in MOX core (wrapper tube),
- insulation of the hot duct in the coaxial piping.

Other GFR-related technologies

- Helium purification system for gas circuits to limit activated impurities and corrosion.
- Tritium management.
- Regeneration of filters.
- Helium recovery from the nitrogen guard vessel gas (helium economy).
- Sealing technology (goal to reduce the leakage rate to 10 % of He inventory per year).
- Wear resistance of materials, especially ceramic thermal shields and insulation.
- Management of helium leaks to containment.
- Qualification of GFR-specific components including material issues, especially
  - gas/gas heat exchangers – design and heat transfer coefficients,
  - active and passive valves (reliability and ageing),
  - fuel handling system,
  - control rods – absorber & cladding material, drive mechanism,
  - reflector around the core – qualification of the material, mechanical & thermal properties,
  - instrumentation (e.g. optical measurements of temperature).

The above list is indicative only; there are other issues that need further attention such as e.g. core physics (voiding reactivity effect, qualification of neutron leakage through axial gas channels, neutronics/thermal hydraulic coupling).

Some phenomena are still not modelled today:

- sub-assembly, core and collector thermalhydraulics,
- core mechanics (equilibrium and seismic response),
- transport of contamination.

4.3 Transmutation in sub-critical reactors

In the past, a range of sub-critical hybrid systems for incineration of spent nuclear fuel have been proposed. At present, only one such project remains, the Belgian MYRRHA Project

(Multipurpose Hybrid Research Reactor for High-tech Applications). MYRRHA has been selected as the ADS track in the SNETP Strategic Research Agenda (SNE-TP SRA 2013).

Objectives

The first objective of the MYRRHA project proposed by SCK-CEN is to establish a multi-purpose research facility serving as a flexible fast spectrum irradiation tool in support of technology development (in particular for materials, components and fuel irradiation tests) of the three fast reactor systems (SFR, LFR and GFR). Also, MYRRHA aims at offering a wide range of irradiation conditions for fusion material research. MYRRHA will be conceived as an Accelerator Driven System, able to work in critical as well as subcritical mode.

As an ADS, MYRRHA contains a proton accelerator of 600 MeV, 4 mA, a spallation target and a multiplying core with MOX fuel, cooled by liquid Lead-Bismuth Eutectic (LBE). The main objectives of this phase of the MYRRHA project is to demonstrate the ADS concept at a reasonable power level on the one hand and, on the other hand, to show the technical feasibility of transmutation of minor actinides in sub-critical systems.

Since MYRRHA is based on heavy liquid metal technology, it couls also contribute to the development of lead fast reactor (LFR) technology. MYRRHA has been adopted as European technology pilot plant (ETPP) in the SNETP roadmap for LFR.

Present status

Following the decision of the Belgian Government to support the MYRRHAproject, SCK-CEN has set up a project structure and team integrating the design and R&D efforts expended in several Seventh Framework Programme (FP7) projects like the CDT (Central Design Team), SEARCH (Safe ExploitAtion Related CHemistry for HLM reactors), MAX (MYRRHA Accelerator eXperiment), MAXSIMA (Methodology, Analysis and eXperiments for the "Safety In MYRRHA Assessment), THINS (Thermalhydraulics of Innovative Nuclear Systems), FREYA (Fast Reactor Experiments for hYbrid Applications), LEADER (Lead-cooled European Advanced DEmonstration Reactor), SARGEN IV (Safety Assessment for Reactors of GENeration IV), SILER (Seismic-Initiated events risk mitigation in LEad-cooled Reactors), MATTER (MATerials TEsting and Rules), GETMAT (Generation IV and transmutation materials), ANDES (Accurate Nuclear Data for Nuclear Energy Sustainability), and HELIMNET (HEavy Liquid Metal NETwork). During the 2010–2014 FEED (Front-End Engineering Design) period the following items should be accomplished:

- primary system and plant design and the associated R&D programme,
- pre-licensing process,
- set-up of the international consortium.

SCK-CEN will be responsible for the primary system, but all other systems, structures and components together with the plant layout has been subcontracted to an international industrial consortium (called FEED-engineer) consisting of Areva and Ansaldo. The Belgian government has assigned 60 M€ to this phase of the MYRRHA project, whereas no international partners have yet joined the project.

For the design of MYRRHA, a strategy has been to benefit as much as possible from previous fast reactor programmes to relax the licensing process. The objective of MYRRHA is also to excel in safety by practically eliminating Fukushima-accident initiators by means of redundant and diversified fully passive decay heat removal systems. Special attention will also be given to design choices and measures for prevention and hence practical elimination of severe accident scenarios.

MYRRHA is a pooltype ADS having its primary and secondary systems designed to evacuate a maximum core power of 100 MWth. All the MYRRHA components are optimised for the extensive use of the remote handling system during component replacements, inspection and handling.

The accelerator is the driver of MYRRHA since it provides the high energy protons that are used in the spallation target to create neutrons which in turn feed the subcritical core. The accelerator will provide a proton beam with an energy of 600 MeV and an average beam current of 4 mA, resulting in a beam power of 2.4 MW.

The planned reactor core consists of mixed oxide (MOX) fuel pins, typical for fast reactors. The requested high fast flux has been obtained by optimising the core configuration geometry (fuel rod diameter and pitch) and maximising the power density. The use of lead-bismuth eutectic (LBE) as coolant permits lowering of the core inlet operating temperature (down to 270 °C) decreasing the risk of corrosion. In subcritical mode the spallation target assembly, located in the central position of the core, brings the proton beam via the beam tube into the central core region. The assembly evacuates the spallation heat deposited there, presents a barrier between the LBE and the reactor hall and assures optimal conditions for the spallation reaction.

The assembly is conceived as an In-Pile-Section (IPS) and is easily removable and replaceable. The encumbrance of the core with the proton beam, the fact that the space situated directly above the core will be occupied by lots of instrumentation and IPS penetrations and the core compactness result in insufficient space for fuel handling to load/unload the core from above. Since the very first design of MYRRHA, fuel handling is thus performed from underneath the core.

The major technological issues for the MYRRHA demonstrator are:

- lead-bismuth chemistry control and conditioning,
- lead-bismuth component testing and thermohydraulics,
- lead-bismuth instrumentation,
- material qualification,
- driver fuel qualification,
- coupling technology of accelerator with subcritical core,
- high intensity proton accelerator performances and reliability.

Challenges

Lead-Bismuth chemistry control and conditioning

For long-term operation of an LBE-cooled ADS, chemistry monitoring and control are crucial for the reactor. An LBE chemistry control and conditioning R&D programme involves the technology related to chemical control of the coolant and purification of the evaporated elements that have low retention in LBE such as Hg. Several issues have been identified for this programme: the development of oxygen sensors to measure the dissolved oxygen concentration in the coolant, the conditioning of the LBE to minimise dissolution of structural materials and core internals and to prevent formation and precipitation of oxides, filtration and trapping of impurities in the LBE, the evaporation and capture of volatile and/or highly radiotoxic elements (e.g. Po-210) from the cover gas and finally the removal of LBE or dissolved constituents from among other components and test samples.

Lead-bismuth component testing and thermal-hydraulics

In order to secure safe and reliable operations of MYRRHA, an extensive R&D programme has been set up to develop and test reactor components. Since MYRRHA is an experimental reactor, fuel handling is a rather frequently occurring task. Two fuel handling machines will be used, based on the rotating plug concept. A high level of reliability of these machines is crucial. Further (thermal-hydraulic) analysis of the fuel assemblies, the MYRRHA core (including control and safety rods), the spallation target and reactor pool is needed to assure long-term operation of the liquid metal cooled MYRRHA reactor. Within the research and qualification programme of the LBE components, the proper working of the primary heat exchangers and the primary pumps must be confirmed.

In-service inspection in Lead-bismuth

The use of LBE as coolant in MYRRHA has also some known disadvantages: its opacity complicates maintenance and fuel handling operations. Developments regarding ultrasonic techniques must be made to improve these operations.

Material qualification

Based on available data on mechanical and thermal properties, irradiation performance, manufacturing and availability, the following steels have been selected as the candidate materials for the components:

- titanium stabilised austenitic stainless steel 15-15Ti for e.g. the fuel cladding of the first cores,
- ferritic-martensitic steel T91 for e.g. the spallation target window,
- austenitic stainless steel type 316L solution annealed for the majority of other components including the reactor vessel, the heat exchangers, the diaphragm and the core barrel.

The austenitic stainless steels including 316L and 15-15Ti have been extensively used in construction of fast sodium cooled reactors in Europe, US and Japan and are therefore relatively well characterised for nuclear applications. However, the innovative nature of the MYRRHA installation poses new challenges for material performance, particularly because of the lead-bismuth eutectic coolant which could be quite corrosive under certain conditions and also might affect the mechanical properties.

The efforts are distributed over the following five overlapping activities:

- identification of key material issues for design and licensing of MYRRHA,
- development of test and evaluation guidelines for characterisation of structural materials,
- assessment of material properties,
- development of testing infrastructure,
- qualification of the chosen materials for the MYRRHA conditions.

Fuel qualification

At present stage of definition of the fuel R&D programme, attention is mainly paid to the driver fuel and cladding material. MYRRHA will rely on conventional fast reactor MOX fuel technology developed and demonstrated in previous sodium programmes like SNR-300, (fuel licensing process only) RAPSODIE, Phénix and Superphénix. The feedback from these programmes covers in many aspects the operating conditions of MYRRHA fuel. Return of experience is maximised and licensing needs are minimised further by choosing the Phénix fuel pin design and cladding material (15-15 Ti) as the preferred option. However, since MYRRHA will have LBE as coolant the topics of clad-coolant interaction and fuel-coolant compatibility have not been dealt with in sodium programmes and are embedded more extensively in the MYRRHA R&D programme. Innovative MA bearing fuels will be loaded in MYRRHA to allow for a further screening and down-selection of these new types of fuels and to finally allow qualification of these innovative fuel types.

Coupling technology of the accelerator with subcritical core

In an Accelerator Driven System the coupling of the accelerator, the target and the subcritical core deserves special attention. The reactor physics of such a coupled system is significantly different from a critical system and dedicated experiments are needed. More specifically, the accurate on-line monitoring of the subcriticality level needs to be validated. To respond to this question, the accelerator-coupled GUINEVERE (Generator of Uninterrupted Intense NEutrons at the lead VEnus REactor) experiment was conceived and validation experiments will need to be carried out. For the coupling of the accelerator and the subcritical core, the beam window serves as a barrier. A dedicated programme on the qualification of the beam window under different MYRRHA conditions is being carried out.

Based on the feedback from the MEGAPIE experiment and the MYRRHA operating characteristics for the beam window, the licensing approach will be based on a maximum supposed lifetime of the beam window of one cycle (3 months operation). In the beginning, the window will be replaced after every cycle while a qualification programme will run in parallel during the first cycles to demonstrate the longer lifetime of the beam window.

High intensity proton accelerator performances and reliability

For MYRRHA a 600 MeV linear proton accelerator with a nominal design current of 4 mA is envisaged. Linear accelerators of this type have been constructed in the past. However, the reliability requirement for the MYRRHA Linear Accelerator is more than one order of magnitude more stringent than what is commonly achieved in research accelerators. Preliminary analyses have shown that in principle the required reliability level should be feasible. However, the realisation of the goal makes a research programme on the accelerator indispensable. The accelerator R&D programme is focussing on:

- injector developments,
- main Linac component developments,
- global accelerator design,
- system optimisation.

In collaboration with UCL (Université Catholique de Louvain), the first part of the MYRRHA accelerator will be built and tested. This injector part, consisting of an ion source, low energy beam transfer line, a 4-rod based Radio-Frequency Quadrupole (RFQ) and a diagnostic section, will deliver protons of 1.5 MeV. The aim is to test first the MYRRHA injector section to analyse and if needed improve its reliability in view of the overall reliability targets. To allow the implementation of a fault-tolerance capability, which is of crucial importance for reliability enhancement, prototypes of the different Linac components will need to be constructed. It is planned to carry out reliability tests by means of prototypes of each cryomodule 'family':

- the SuperConducting Cross-bar H-type (SC-CH) cryomodule,
- the spoke cryomodule,
- the long elliptical cryomodule.

Besides the cryomodules themselves, critical components to be developed are:

- the superconducting Radio Frequency (RF) power couplers,
- the Low Level RF (LLRF) required for the fault tolerant scheme.

The Linac design will be consolidated by means especially of advanced beam simulations based on start-to-end simulations and associated error analyses together with assessments on new R&D results, new reliability studies, definition of preliminary infrastructures and revised cost estimates. Recommendations will be formulated, including a roadmap towards the actual construction of the MYRRHA accelerator. Links between the activities of the FP7 project MAX, the results of the FP7 project CDT, and the related R&D ongoing in the accelerator community will be set-up.

During the FP6 EUROTRANS project (European Research Programme for the Transmutation of High Level Nuclear Waste in an Accelerator Driven System), a preliminary reliability study of the ADS reference accelerator has been conducted in order to assess the number of beam trips. Such beam trips threaten the core materials and can affect dramatically the plant availability. It is intended to pursue these reliability-oriented studies and to develop a more accurate reliability model of the MYRRHA accelerator. A model of the full MYRRHA Linac will be built taking into account all support systems and, as far as possible, smart control strategies, fast beam shutdown systems and accelerator/reactor interface aspects.

4.4 Partitioning

The challenges for R&D in fuel reprocessing include

- industrialisation of laboratory technologies for separation of minor actinides from the high level waste from the reprocessing of the fuel used in present reactors,
- development of reprocessing of advanced fuels foreseen for future reactors (FNR, ADS, advanced thermal reactors and HTR),
- technologies able to perform joint extraction of several actinides, and
- minimisation of secondary wastes in all these strategies.

These developments should be performed coherently with the technologies for advanced fuel fabrication and characterisation. National and international initiatives including the ACSEPT project have been devoted to reprocessing and conditioning of LWR and advanced fuels for MA separation. In particular that project has provided progress on the definition of extraction molecules and the concept for hydrometallurgical process with MA separation. Research is underway to study the stability of these molecules under realistic radiological and chemical conditions and to assess the industrialisation of the process. (SNE-TP SRA 2013).

On the pyro-metallurgical side, the activites are more of a fundamental R&D character. In parallel, some national studies and others involving several EU countries continue developing the processes for dissolution of MA-bearing MOX and MA targets using a different basic matrix. In the short term, the required R&D for nuclear waste reprocessing can be performed in several existing basic science and validation facilities, but in the medium term demonstration plants for the reactors, fuel fabrication and advanced reprocessing technologies will be needed, both at national and European (joint) levels.

The priorities for short-term R&D in fuel reprocessing are:

- advanced reprocessing of LWR and advanced fuels for MA separation, using either hydro- or pyrometallurgical processes,
- dissolution of MA-bearing MOX and carbide fuels for FNRs and of MA bearing targets (U-free or UO_2 matrix),
- conversion processes after the separation steps and prior to the re-fabrication of fuels/targets,
- processes for HTR fuel recycling and waste reduction, integration of fuel cycle with LWR and FNR,
- synthesis of new fuels and their performance assessment, oriented to their reprocessing,
- irradiation behaviour of MA-bearing MOX and carbide fuels, and MA bearing targets and dedicated PIE programmes.

In the medium term the R&D will need demonstration facilities. The decision to develop or not demonstration facilities for fuel fabrication and reprocessing should be taken in about 2017 depending on the results of the previous steps and of the availability of equivalent facilities in Europe. Within its programme to operate a sodium cooled fast reactor prototype by 2020, France is considering the construction of a facility devoted to the manufacturing of the core fuel for the ASTRID FR prototype. This facility, called AFC, could also provide fuel fabrication services for the testing and demonstration of alternative reactor technologies at the European level.

For recycling ASTRID fuel, several options are under consideration: either a dedicated pilot-scale facility (ATC), or adaptation using complementary steps in a LWR fuel reprocessing plant. Meanwhile an investigation is under way in order to evaluate the possibilities of increasing the capacities of existing facilities such as ATALANTE (CEA/Marcoule). Demonstrative transmutation experiments, at sub-assembly scale, would call for new facilities, able to manufacture MA-bearing targets or MA-bearing fuels, at kg scale. The objectives of these facilities should be to prepare the next generation reprocessing plants, which will be needed for the fast reactor cycle. They should address issues such as multi-recycling of plutonium and minor actinide separation.

In the long term, the R&D should focus on the industrial implementation of partitioning and transmutation. The implementation of this phase will depend on the results of the previous phases and will be mainly carried out under the control of the nuclear industry. Non-MOX fuel still needs some reprocessing development.

As mentioned earlier the pyro-processing is not as far come as the hydro processing and therefore the aim of the coming years is very different. There is still a great need to obtain data for possible different fuels and separation routes. Whether it will concern molten salt/molten metal separation or electrochemical separation does not matter. Currently there is no real reason to select one or the other, which is why efforts should be made in both directions to obtain enough flexible data and modelling ability to make pyro-processing able to face the uncertainties of future high burn-up fuels with possibly high minor actinide content.

There is also a materials problem that needs to be addressed in the future. The wear and tear of pyro-processing equipment is high and even if knowledge is available from the mining industry etc specific demands will be raised for materials used to process molten spent nuclear fuel. This is, however, a subject that is not typically addressed by nuclear chemists but highly skilled metallurgists that will have to join this part of process development.

4.5 Nuclear data

4.5.1 Future work for the UU group

The main steps to be taken by the applied nuclear physics division in relation to partition and transmutation research seem currently very clear:

- improvement of the local infrastructure for detector development activities in Uppsala,
- conduction of the planned activities at NFS, Caen, France and IGISOL, Jyväskylä, Finland,
- continuation of the work on the TMC method including development of the TALYS code,
- continuation of the work on safety and security aspects in Gen IV systems.

The local infrastructure has already started to be improved since a new laboratory for detector development has been taken into operation late 2012. The division is also constructing a small 14 MeV neutron beam facility on the Ångström Campus site for detector testing and small size experimental studies. This facility is planned to be taken into operation during autumn 2013.

The planned experimental activities at IGISOL should start during 2013. At first proton induced fission studies will be performed to test the new beam line. Installation of the neutron production target to fast reactor like spectra and first test runs should talk part during autumn. The NFS activities will start in 2014 at the earliest. Meanwhile the developed detectors will be tested with neutrons from a Cf-252 source and at the planned 14 MeV neutron beam.

The computational work with the TMC approach is expected to play an important role in the division's future activities. Arjan Koning from NRG Petten, the main developer of the TALYS code and one of the two key persons behind the TMC method has been closely attached to UU by an adjoint professorship.

Among many promising research tasks UU will focus on improving the theoretical modelling of fission yield data with TALYS, in order to improve fission yield predictions and make them available for application in reactor and inventory calculations. For this activity a new postdoc will be hired. This will improve TMC capabilities for studying Gen IV fuel cycles, especially investigation of the impact of nuclear data uncertainties, in particular uncertainties of fission yields, on the fuel cycle for thermal (especially HTR) and a fast system (e.g. ASTRID and ELECTRA). Our involvement in TENDL (Koning et al. 2013) will continue and methods for optimizing nuclear data libraries with respect to fast spectrum reactors will be a prioritized research task. Furthermore it is planned to investigate how differential and integral measurements affect the uncertainty of selected isotopes. As a starting point Fe and Pb isotopes will be studied. For this activity it is planned to hire a new PhD student. Last not least it would be desirable to test the TMC method with deterministic reactor codes.

Gen IV research will be extended by studies concerning the possibilities to use today's nuclear waste as fuel in future Gen IV reactor systems. As an example, one may mention research on so-called blankets that are suggested in some Gen IV concepts, of which the ASTRID prototype reactor is one. In such a configuration, the core would comprise fertile zones with a content of, e.g., depleted uranium, in which fissile material can be bread to sustain the system for longer times. However, such blankets cause concerns from a non-proliferation perspective, since they may involve the production of weapons-grade plutonium. Research performed by the nuclear fuel and safeguards group suggests the use of spent LWR fuel (UOX or MOX) in the blanket, which would cause a degradation of the plutonium present in the blanket.

Besides the above mentioned steps, the nuclear reactions group works on identifying new collaborations for experimental studies. In particular contacts with the nTOF collaboration at CERN, Switzerland, have been taken up concerning possible measurements of elastic scattering at the EAR-2 beam line currently under development. This beam line is planned to be put in operation late 2014 and should, due to shorter distance to the neutron production target than the main nTOF beam line, offer sufficient intensity for this kind of measurement. Elastic scattering cross sections are both of key importance for nuclear model code development and often used as reference cross section for other measurements. The nTOF beamline would allow measuring especially the elastic *np* cross section over a wide energy range.

4.5.2 EU projects – CHANDA

Following up on, in particular, the ANDES and ERINDA projects, a collaboration of 36 partners lead by CIEMAT, Spain, has submitted a proposal to the EC within the FP7 program on fission. The proposal carries the title "Solving Challenges in Nuclear Data for the safety of European nuclear facilities", CHANDA, applying for a EC contribution of 6 M€. At present, there are a number of challenges to the nuclear data to reply to all the needs associated to the improvement of the safety of nuclear systems:

1. To optimize the use of resources, facilitating the access of experimental teams to access the available EU nuclear data facilities (i.e. transnational access, similar to the EFNUDAT and ERINDA project, to facilities like TSL in Uppsala).
2. To build facilities that provide much larger number of neutrons for differential measurements of nuclear data to allow measurements on short lived and rare materials (in particular aiming at NFS in Caen, France and the EAR-2 beamline at nTOF, CERN, Switzerland).
3. To be able to fabricate the targets needed for the experimental measurements, particularly radioactive targets and with special attention to actinides and fission fragments.
4. To develop the methodologies and capabilities of the facilities and the R&D teams performing measurements, evaluation and validation of nuclear data and nuclear models.

The work is proposed to be split into several domains and UU is involved in several projects in several of the domains; UU contributes to the development of the NFS facility in France, measurements of fission yields at IGISOL, Jyväskylä, Finland, and the development of the predictive power of TALYS nuclear model code. The later is of key importance for uncertainty and covariance assessments. Furthermore, TSL is part of the domain on transnational access to facilities.

4.6 Interfacial radiation chemistry

The planned research at the division of applied physical chemistry, KTH Royal Institute of Technology, can be divided in two parts: (1) Radiation induced dissolution of new fuel types (mainly nitride fuel) and (2) Radiation induced corrosion of metallic materials and interfacial radiation chemistry. In parallel with these two research tracks, a numerical model for simulation of radiation induced dissolution of spent nuclear fuel and radiation induced corrosion of metallic materials will be developed. As this is a rather extensive research program, internal as well as external (national and international) collaborations are essential. The different parts of the program are described in more detail below.

4.6.1 Radiation induced dissolution of new fuel types (mainly nitride fuel)

The mechanism and kinetics for radiation induced dissolution UO_2-based nuclear fuel have been studied for more than a decade. During this period we have contributed to the understanding of the mechanism and kinetics for oxidative dissolution of UO_2 in water containing HCO_3^-, the mechanism and kinetics for the hydrogen effect on oxidative dissolution of spent nuclear fuel and developed a fairly simple model describing the dynamics of radiation induced dissolution of UO_2-based nuclear fuel, accounting for the hydrogen effect (Jonsson et al. 2007). We intend to use the same methodology in our planned studies of new fuel types of relevance in a nuclear fuel cycle based on transmutation. A central part the project is a comparison of the results on new fuel types with previous results on UO_2-based fuel. In short, the following types of experiments will be carried out:

H_2O_2 induced oxidative dissolution in aqueous solution.

H_2O_2 has been shown to be the most important oxidant in radiation induced dissolution of UO_2. For this reason it is important to study the reactivity of H_2O_2 towards other types of fuel matrices and to what extent these reactions result in matrix dissolution.

Mechanistic studies of reactions between H_2O_2 and fuel surfaces.

H_2O_2 can react with UO_2 in two different ways: (1) A redox reaction resulting in oxidation of UO_2 and (2) Catalytic decomposition of H_2O_2 on the UO_2-surface. The first reaction leads to oxidative dissolution of the matrix while the other reaction does not affect the matrix. It is essential to know the distribution between the two possible reaction pathways. For this reason we have developed a method where the primary product of the catalytic decomposition (hydroxyl radical) can be quantified.

Studies of γ-radiation induced dissolution of fuel pellets and powder.

Studies of γ-radiation induced dissolution of pellets and powder are important as a complement to the studies described above. From the radiation chemical experiments (in combination with numerical simulations of the same system) we can assess the relative impact of the different radiolysis products.

Surface characterization after exposure to H_2O_2 and after irradiation.

To get the full picture of the processes involved when the fuel matrix is irradiated or exposed to H_2O_2, careful characterization of the surfaces prior to and after irradiation or exposure to H_2O_2 is required. We mainly plan to use SEM but Raman, XRD and XPS will also be employed.

Impact of solutes

Groundwater contains several constituents that can influence oxidative dissolution of the fuel matrix. The most important constituent in oxidative dissolution of UO_2-based fuel is HCO_3^-/CO_3^{2-} which forms soluble complexes with UO_2^{2+} and thereby enhances the rate of release of the oxidation product from the fuel surface. This enhances the overall process and it is therefore important to quantify the concentration dependence. Other components such as iron (Fe^{2+}) and sulfide (HS^-) act as reductants and can thereby slow down the whole process.

Effects of epsilon particles and hydrogen.

Noble metal particles (composed of fission products) formed in the fuel matrix during the fission process have been shown to be very efficient catalysts for H_2-reduction of U(VI) on the surface of the matrix. This process has been shown to be the main route responsible for H_2-inhibition of radiation induced dissolution of nuclear fuel. It is crucial importance to investigate the impact of this process on the dynamics of the dissolution of other fuel types.

Effects of fission products in the fuel matrix (compare to SIMFUEL)

Several fission products form oxides that are incorporated into the fuel matrix. For UO_2-based fuel it has been shown that the presence of rare earth metal oxides in the UO_2-matrix strongly influences the matrix reactivity towards oxidants. In general, the fission products protect the matrix from being oxidized. The effect of fission products should be investigated also for other fuel types.

The types of experiments described above are described in detail in refs. Eriksen et al. (2012), Hossain et al. (2006), Jonsson et al. (2007), Nilsson and Johnsson (2011), Roth and Johnsson (2008, 2009), Trummer et al. (2008, 2010) and Trummer and Jonsson (2010). A significant fraction of the work will be carried out in collaboration with Nuclear Chemistry at Chalmers and Reactor Physics at KTH.

4.6.2 Radiation induced corrosion of metallic materials and interfacial radiation chemistry.

Interfacial radiation chemistry in general and radiation induced corrosion of metallic materials in particular are of paramount importance in all nuclear technological applications. Nevertheless, data on these processes are scarcely reported in the scientific literature. In this field we lack basic knowledge about the fundamental processes as well as knowledge about more specific systems. In the framework of the present project we intend to systematically gain knowledge about the primary processes and apply this knowledge on more specific systems. Since a couple of years we study radiation induced corrosion of copper. Within the present project we intend to study radiation induced corrosion of steel. The methodology we will use is essentially the same as has been used to study copper (Björkbacka et al. 2012). The following types of experiments will be performed:

Radiation induced dissolution of metal ions.

To obtain the full picture of radiation induced corrosion of steel (or other materials) it is necessary to study the release of metal ions to the liquid phase and to characterize the surface. In this part of the project we will use ICP-OES to determine the rates of oxidative dissolution of steel.

Surface characterization before and after irradiation

As previously mentioned, surface characterization is essential for the understandingo f the process. We will use SEM, Raman, XRD, XPS, AFM and IRAS to characterize the surfaces before and after irradiation. In addition, electrochemical techniques will be used to determine oxide layer thickness.

Electrochemical studies during irradiation

Electrochemical techniques can provide valuable information about corrosion processes. We intend to develop methodology to perform electrochemical studies during irradiation. The results from such measurements will be compared with experiments carried out without irradiation in solutions containing well controlled concentrations of oxidants. This work will be carried out in collaboration with the division of surface and corrosion science at KTH.

Mechanistic studies (for individual radiolysis products)

In the same manner as for different fuel types, we intend to study the relative impact of individual radiolysis products on the corrosion of steel. H_2O_2 will be the first candidate and we will study the kinetics for the reaction between H_2O_2 and steel as well as the mechanism for this reaction. Catalytic decomposition of H_2O_2 on the steel surface will also be studied.

Studies of other materials than steel

To better understand the effects of the composition and structure of the solid material we will use other materials than steel as reference materials. This includes pure metals as well as alloys. The same types of studies as described above will be carried out for these materials.

Fundamental studies of interfacial radiation chemistry will be carried out through international collaborations with University of Notre Dame (Prof. Jay LaVerne), Brookhaven National Laboratory (Dr. Robert Crowell), University of Manchester (Prof. Simon Pimblott) and University of Tokyo (Prof. Yosuke Katsumura). Here it is important to understand interactions between radiolysis products and oxide surfaces and to what extent the presence of an oxide surface influences the primary yields of radiolysis products. We have already carried out some studies in this field (Lousada and Jonsson 2010, Lousada et al. 2012).

Development of a numerical model for simulation of radiation induced surface reactions.

The purpose of the first two parts of the project is to provide fundamental knowledge concerning the mechanism and kinetics for processes relevant in radiation induced corrosion or dissolution. This information should be used to predict the life time a material in a given environment (e.g. deep repository for spent nuclear fuel, nuclear reactor or reprocessing plant). To achieve this, simulations based on models including relevant processes and parameters are necessary. Such models are available for homogeneous radiation chemistry in aqueous solutions and there are also models describing corrosion processes. However, there is no model connecting interfacial radiation chemistry and corrosion. We intend to develop such a model in the framework of this project.

5 Conclusions

5.1 Impact on nuclear energy and waste programme

Successful development of P&T within an advanced nuclear fuel cycle will not make a deep geological repository obsolete. The complex processes will inevitably generate some waste streams with small amounts of long-lived radionuclides.

Successful development of P&T may, however, decrease the requirements on in particular the engineered barriers as well as on the volumes required in a deep geological repository. A recent NEA-study (NEA 2006a) indicates for example that the volume requirements in a repository for high level waste (per unit of electricity produced) may be of 5 to 20 times more for electricity from light water reactors with once through fuel cycle than from advanced reactors with advanced fuel cycle processes. On the other hand the amounts of long-lived low and intermediate level wastes will be larger for fuel cycles with recycling of transuranium elements (NEA 2006a, pp 67–70).

For Sweden it is important to participate in the international development and to maintain a reasonable level of competence in the country. Competence developed in research on advanced fuel cycles and waste management strategies including P&T is valuable and useful also in the work on maintaining and developing safety and fuel supply for the existing light water reactors. It is also of importance for assessing the further development of the waste management programme.

The implementation of partitioning and transmutation to effectively reduce the amount of long-lived radionuclides that must be placed in a geological repository necessitates a commitment to nuclear power for a very long time, at least more than 100 years.

About a decade ago, some anti-nuclear groups advocated P&T with ADS as a “clean-up” mission. This is no longer the case; few sceptics today are P&T proponents. The general conclusion today that P&T requires century-scale implementation to make a significant difference on the radio-toxicity of spent nuclear fuel has made P&T a less attractive solution in the eyes of the anti-nuclear movement.

5.2 Acceptance

Advanced fuel cycles and waste management strategies including P&T on an industrial scale require large nuclear facilities that must be accepted and trusted by society. As such strategies involve very long lead times the trust and acceptance requirements become very strong. This means that the benefits must be perceived as very large and important.

5.3 Partitioning

The major development efforts on partitioning are still concentrated upon aqueous processes. Pyrochemical separation processes attract considerable interest, but they may be reserved for special types of fuel with very highly radioactive contents.

The progress made in partitioning is promising. The goal to get a separation efficiency of 99.9 % of the actinides from other elements in the spent fuel seems to be possible to reach.

A problem with the aqueous partitioning processes so far is the cost and stability of extraction agents.

Extrapolation of an aqueous partitioning process to industrial scale is rather straightforward once the flow sheet has been defined and tested on laboratory scale.

Pyrochemical processes needs further development. Before such processes can be extrapolated to industrial scale new types of equipment will require full scale tests with inactive and radioactive material.

A feasibility and systems study of the partitioning processes covering technical but also proliferation, cost and safety aspects might be useful for guiding the further development in this area.

5.4 Transmutation

There is broad agreement that effective transmutation/burning of all the transuranium elements can only be achieved in fast neutron spectra.

The main interest for transmutation has shifted back to critical fast reactors. For some years it was largely focused on accelerator-driven systems (ADS). These are now seen as an option for burning minor actinides in a second stratum, if used at all, where LWR and FR constitute the first stratum.

There is no consensus on the basic technical components of a full scale P&T facility, neither concerning the coolant nor the fuels. There seems to be some agreement that liquid sodium cooling is most technically mature and therefore closest to deployment. Lead cooling is argued to allow a principally simpler technical system, given that the corrosion and pumping problems can be solved.

When it comes to test facilities, two clear routes are present in Europe. ADS, that previously attracted considerable attention, is now limited to one single project, the Belgian Myrrha concept. This has been identified by SNE-TP as the preferred choice for such a test facility (SNE-TP SRA 2009), and the Belgian government has recently assigned 60 M€ to next phase of the project. Matching funding from other countries or the EU is still missing though.

With adequate funding, it would take another five to ten years to construct the facility, followed by development and construction of an almost full-scale demonstration plant that might be completed in the middle of the 2030ies. The funding levels of the R&D-programmes on ADS have, however, not reached the levels required to meet this time target. It is thus unlikely that the time schedules suggested in the ADS-systems studies will be met.

The previous prediction that deployment of ADS-plants on an industrial scale is unlikely to occur before 2050 (Ahlström et al. 2004, 2007, Blomgren 2010) is still valid, but the reasons have shifted over time. Previously, this prediction has been based on the technical difficulties in combination with the relatively modest funding level. Our present prediction is rather based on the increased attention on FRs, that can be expected to have two side-consequences. First, FR development can be expected to solve many of the technical problems involved in ADS, in particular materials and fuel issues. Second, deployment of FR will most likely have higher priority than ADS development. Moreover, the final performance of fast reactors will be of seminal importance for the design of ADS. It seems generally accepted now that if used at all, ADS should constitute a possible second stratum, with minor actinide incineration as main rationale. Thus, the amounts and composition of the waste streams from FR are of crucial importance for the ADS design. It seems therefore reasonable that ADS research is performed keeping a range of options open until the performance of FRs is clarified, and detailed design of an industrial ADS can thereby be expected to await an initial FR deployment and operation phase.

Fast reactors can be used either as burners of plutonium and other transuranium elements or as breeders for most efficient use of the mined natural uranium. France has announced the intent to construct a modern sodium-cooled fast reactor, ASTRID, by 2022. This project has been adopted as the first step in the SNE-TP Strategic Research Agenda (SNE-TP SRA 2009). In the end of 2012, a conceptual design was released and submitted to the French Parliament for assessment. The aim is to take a decision in summer 2013 about whether to proceed. Most predictions today implicate that a positive decision will be taken, aiming at full design by 2017.

Both ADS and fast reactors need improved fuel, improved material and other improved equipment before deployment on an industrial scale. New fuel, new material and new equipment will need full scale testing before being used in an industrial plant.

5.5 Safety and radiation protection

The implementation and construction of future advanced nuclear fuel cycle (including P&T) facilities must meet the same requirements from environmental and nuclear laws as other nuclear facilities.

The development of such facilities involves the resolution of many issues concerning safety and radiation protection. As the systems are under basic development it is unlikely that all such issues have been identified.

5.6 Time and costs

As is indicated above, the development of any P&T-system will require several decades. The deployment and operation of the necessary industrial facilities will take another 100 years or more before the stocks of long-lived radionuclides from the currently existing nuclear power reactors have been transmuted.

The costs for P&T are not possible to assess with reasonable confidence at the current stage. New assessments that have been published during recent years are all based on the same data as those presented in the previous status reports (Ahlström et al. 2004, 2007, Blomgren et al. 2010). The new results do not change the picture presented there.

The efficient use of the energy released by the transmutation process is essential for any possibility to achieve economic viability for P&T. This is particularly the case for transmutation of plutonium. Some experts have the view that minor actinides might be transmuted separately in dedicated ADS-plants where the generated heat could be dumped, but this represents a minority view. Most experts believe society will not take the costs for incineration without reducing the penalty by utilizing the heat for electricity production.

It seems that for future large scale nuclear energy production a system with light water reactors may dominate for a considerable time. All nuclear new-build in Europe, and essentially all in the World, is based on light-water reactors, and they can be expected to operate minimum 60 years, and 80 or even 100 years is not out of question. The light-water reactor backbone may over time be supplemented by fast reactors for burning of plutonium and perhaps by ADS-facilities for transmutation of other transuranium elements as well as of some plutonium with high content of heavy isotopes.

In the long term, the uranium prices may increase due to depletion of the low-cost resources or due to demands for very strict environmental measures in the mining. Then fast breeder reactors may take over the role as leading nuclear power producers. Such reactors are also feasible for transmutation of the heavier transuranium elements. At present, it seems likely that a majority of the nuclear power reactors in the World are light-water reactors for the rest of this century.

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